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SUBJECT: Application for amends to Licenses DPR-58 & DPR-74,changing Westinghouse fuel & reload analysis methodology.

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Donald C. Cook Nuclear Plant Units 1 and 2
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74
UNIT NO. 2 CYCLE 8 RELOAD LICENSING, PROPOSED TECHNICAL
SPECIFICATIONS FOR UNIT 2 CYCLE 8, AND RELATED
UNIT 1 PROPOSALS

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Attn: T. E. Murley

February 6, 1990

Dear Dr. Murley:

This letter and its attachments constitute an application for amendment to the Technical Specifications (T/Ss) for Donald C. Cook Nuclear Plant Units 1 and 2. This amendment is requested to support the Cycle 8 reload of Unit 2. Indiana Michigan Power Company will reload the Donald C. Cook Nuclear Plant Unit No. 2, Cycle 8 with Westinghouse Vantage 5 (V5) fuel assemblies. Westinghouse has replaced Advanced Nuclear Fuels Corporation (ANF) as the fuel supplier for Unit 2. The majority of these proposed T/S changes are related to the change to Westinghouse fuel and reload analysis methodology. As discussed below, certain related Unit 1 T/S changes are also proposed. Entry into Mode 4 for Cycle 8 is anticipated to occur on or about August 24, 1990.

Content of the Submittal

This submittal addresses two issues in addition to the proposed T/S changes for the Unit 2 core for Cycle 8. These are:

1) The Unit 2 Licensing Basis

When Unit 2 was relicensed for Cycle 6 operation, a newly revised Exxon Nuclear Company, Inc. (now Advanced Nuclear Fuels Corporation) methodology was employed. This methodology was based on the NRC's Standard Review Plan (NUREG-0800). As a result, seven events not in the Unit 2 licensing basis were analyzed. These seven events will

A001 1/1 Add: NRR/Chatterton IRM Tech Adv Ltr Encl 1 1 1 1

not be analyzed for Unit 2 Cycle 8 or subsequent cycles. This issue was discussed with the staff on June 12, 1989. It is addressed in more detail in Attachment 8.

2) Proposed Changes to the Unit 1 T/Ss

There are a few changes to the Unit 1 T/Ss being proposed. These occur where the justification for a proposed Unit 1 T/S change is essentially identical to the justification for a similar change to the Unit 2 T/Ss. By proposing the change for both units, efficiency is achieved in the review effort. In addition, the T/Ss for the units are maintained more nearly alike.

The Generic Letter 88-16 Submittal

In parallel with this submittal, we are submitting our proposed T/S changes for Unit 2 in response to Generic Letter 88-16, "Removal of Cycle Specific Parameter Limits from Technical Specifications." Our identifier for the Generic Letter submittal is AEP:NRC:1077A. In AEP:NRC:1077A we propose a Core Operating Limits Report (COLR) for Donald C. Cook Nuclear Plant Unit 2. We will need your response to this submittal as soon as possible. If the staff cannot approve our Generic Letter 88-16 submittal, it will be necessary to propose additional T/S changes for Cycle 8 that would have been submitted in the COLR document. These proposed T/S changes would include:

- moderator temperature coefficient (MTC),
- MTC 300 ppm MTC surveillance acceptance criterion,
- all rods out (ARO) position and control rod insertion limits,
- axial flux difference allowable deviation, and axial flux difference target band,
- F_Q and K(Z)
- $F^N_{\Delta H}$ and $F^N_{\Delta H}$ slope

The values of most of these parameters are different from those for Cycle 7. Attachment 7 contains the values of the above parameters currently planned for Cycle 8.

Please advise us before March 31, 1990 of your intentions regarding our COLR submittal so that we can take appropriate action to ensure timely approval of this submittal.

Organization of the Submittal

This submittal is organized to facilitate the reviewer's task. A detailed description of the organization of the submittal is found at the beginning of Attachment 1. This description will direct the reviewer to the locations of the significant hazards consideration analysis, proposed T/S changes, and supporting documentation.

Other Licensing Considerations

1) Environmental Aspects of Extended Burnup Fuel

   The Unit 2 Cycle 8 fresh fuel assemblies will be limited to 4.2 weight percent U-235 and at discharge will not exceed 56,000 MWD/MTU. The environmental aspects of extended burnup fuel were addressed in a previous submittal identified as AEP:NRC:1071F.

2) Feedwater System Malfunctions Causing an Increase in Feedwater Flow

   Review of this non-LOCA accident is continuing. If this review requires any change to the information supplied in Attachment 4, Appendix B, Section B.3.8a.2, the staff will be notified prior to April 15, 1990. We do not anticipate that any change will be required.

These proposed T/S changes have been reviewed by the Plant Nuclear Safety Review Committee and by the Nuclear Safety and Design Review Committee.

In compliance with the requirements of 10 CFR 50.91(b)(10), copies of this letter and its attachments have been transmitted to Mr. R. C. Callen of the Michigan Public Service Commission and the Michigan Department of Public Health.

This document has been prepared following Corporate procedures that incorporate a reasonable set of controls to ensure its accuracy and completeness prior to signature by the undersigned.

Sincerely,

M P Alexich

M. P. Alexich
Vice President

ldp

Attachments

cc: D. H. Williams, Jr.
A. A. Blind - Bridgman
R. C. Callen
G. Charnoff
A. B. Davis - Region III
NRC Resident Inspector - Bridgman
NFEM Section Chief

SAFETY EVALUATION

FOR THE

DONALD C. COOK NUCLEAR PLANT UNIT 2

TRANSITION TO WESTINGHOUSE 17X17 VANTAGE 5 FUEL

9002120256

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1.0 INTRODUCTION AND CONCLUSIONS

Donald C. Cook Nuclear Plant Unit 2 (Cook Nuclear Plant Units 2) is currently operating in Cycle 7 with an Advanced Nuclear Fuels (ANF) core. Beginning with Cycle 8, it is planned to refuel and operate with the Westinghouse VANTAGE 5 improved fuel design except for the inclusion of a Debris Filter Bottom Nozzle instead of the VANTAGE 5 bottom nozzle. As a result, future transition core loadings would range from approximately 40% VANTAGE 5 and 60% ANF to eventually an all VANTAGE 5 fueled core. The VANTAGE 5 fuel assembly was designed as a modification to the current Westinghouse Optimized Fuel Assembly (OFA) design (Reference 1).

The VANTAGE 5 design features were conceptually packaged to be licensed as a single entity. This was accomplished via the NRC review and approval of the "VANTAGE 5 Fuel Assembly Reference Core Report," WCAP-10444-P-A (Reference 2). The initial irradiation of a fuel region containing all the VANTAGE 5 design features occurred in the Callaway Plant in November 1987. The Callaway VANTAGE 5 licensing submittal was made to the NRC on March 31, 1987 (ULNRC-1470, Docket No. 50-483). NRC approval was received in October 1987. Several of the VANTAGE 5 design features, such as axial blankets, Reconstitutable Top Nozzles, extended burnup modified fuel assemblies and Integral Fuel Burnable Absorbers have been successfully licensed as individual design features and are currently in operating Westinghouse plants.

A brief summary of the VANTAGE 5 design features and major advantages of the improved fuel design and the Debris Filter Bottom Nozzle are given below. These features and figures illustrating the design are presented in more detail in Section 2.0.

Integral Fuel Burnable Absorber (IFBA) - The IFBA features a zirconium diboride coating on the fuel pellet surface on the central portion of the enriched UO_2 pellet stack. In a typical reload core, approximately one third of the fuel rods in the feed region are expected to include IFBAs. IFBAs provide power peaking and moderator temperature coefficient control.

Intermediate Flow Mixer (IFM) Grids - Three IFM grids located between the three uppermost Zircaloy-H grids provide increased DNB margin. Increased margin permits an increase in the design basis $F^N_{\Delta H}$ and F_Q.

Reconstitutable Top Nozzle (RTN) - A mechanical disconnect feature facilitates the top nozzle removal. Changes in the design of both the top and bottom nozzles increase burnup margins by providing additional plenum space for fission gas accommodation and room for fuel rod growth.

Extended Burnup - The VANTAGE 5 fuel design will be capable of achieving extended burnups. The basis for designing to extended burnup is contained in the approved Westinghouse extended burnup topical WCAP-10125-P-A (Reference 3).

Blankets - The axial blankets consist of a nominal six inches of natural UO_2 pellets at each end of the fuel stack to reduce neutron leakage. Loading patterns utilizing radial blankets are shown to further improve uranium utilization and provide additional pressurized thermal shock margin.

Debris Filter Bottom Nozzle (DFBN) - This bottom nozzle is designed to inhibit debris from entering the active fuel region of the core and thereby improves fuel performance by minimizing debris related fuel failures. The DFBN is a low profile bottom nozzle design made of stainless steel, with reduced plate thickness and leg height. The DFBN is structurally and hydraulically equivalent to the existing low profile bottom nozzle.

This submittal is to serve as a reference safety evaluation/analysis report for the region-by-region reload transition from the present Donald C. Cook Unit 2 ANF fueled core to an all VANTAGE 5 fueled core. This submittal examines the differences between the VANTAGE 5 and the ANF fuel assembly desings andevaluates the effect of these differences on the cores during the transition to an all VANTAGE 5 core. Although it is anticipated that the Cook Nuclear Plant Unit 2 will be initially operated in Cycle 8 at the currently licensed core power level of 3411 MWt, unless specifically indicated, the VANTAGE 5 core evaluations and analyses were performed to support an uprate to a core thermal power level of 3588 MWt. The following assumptions made in the safety evaluations and analyses: a full power of $F^N_{\Delta H}$ of 1.62 for the VANTAGE 5 fuel and 1.55 for the ANF fuel, maximum F_Q of 2.22 for the VANTAGE 5 fuel and 2.10 for the ANF fuel and a 15% peak and 10% average steam generator tube plugging level.

The approved Westinghouse Revised Thermal Design Procedure (RTDP) is used in the DNB analyses of both VANTAGE 5 and ANF fuel assemblies for all DNB related accidents, excluding transients such as the hypothetical steamline break where RTDP methodology is not applicable. For such transients standard DNB design methods are used. The WRB-2 DNBR correlation is used in the VANTAGE 5 DNB analyses. The ANF fuel is analyzed by using the W-3 DNB correlation.

The standard reload design methods described in Reference 4 and will be used as a basic reference document in support of future Cook Nuclear Plant Unit 2 Reload Safety Evaluations (RSEs) for VANTAGE 5 fuel reloads. Sections 2.0 through 5.0 summarize the Mechanical, Nuclear, Thermal and Hydraulic, and Accident Evaluations, respectively. Section 6.0 gives a summary of the changed needed to the Technical Specifications. Appendices A and B contain the Technical Specification change pages and non-LOCA safety analyses results, respectively. Appendix C contains the large and small break LOCA safety analyses.

Consistent with the Westinghouse standard reload methodology, Reference 4 parameters are chosen to maximize the applicability of the safety evaluations for future cycles. The objective of subsequent cycle specific RSEs will be to verify that applicable safety limits are satisfied based on the reference evaluation/analyses established in this RTSR.

In order to demonstrate early performance of the VANTAGE 5 design product features in a commercial reactor, four VANTAGE 5 demonstration assemblies (17x17) were loaded into the V. C. Summer Unit 1 Cycle 2 and began power production in December of 1984. These assemblies completed one cycle of irradiation in October of 1985 with an average burnup of 11,357 MWD/MTU. Post-irradiation examinations showed all 4 demonstration assemblies were of good mechanical integrity. No mechanical damage or wear was evident on any of the VANTAGE 5 components. Likewise, the IFM grids on the VANTAGE 5 demonstration assemblies had no effect on the adjacent fuel assemblies. All four demonstration assemblies were reinserted into V. C. Summer 1 for a second cycle of irradiation. This cycle was completed in March of 1987, at which time the demonstration assemblies achieved an average burnup of about 30,000 MWD/MTU. The observed behavior of the four demonstration assemblies at the end of 2 cycles of irradiation was as good as that observed at the end of the first cycle of irradiation. The four assemblies were inserted for a third cycle of irradiation which was completed in November of 1988 (EOC burnup

46,000 MWD/MTU). Post-irradiation examinations showed all four assemblies were still in good mechanical condition.

In addition to V. C. Summer, individual VANTAGE 5 product features have been demonstrated at other nuclear plants. IFBA demonstration fuel rods have been irradiated in Turkey Point Units 3 and 4 for two reactor cycles. Unit 4 contains 112 fuel rods equally distributed in four demonstration assemblies. The IFBA coating performed well with no loss of coating integrity or adherence. The IFM grid feature has been demonstrated at McGuire Unit 1. The demonstration assembly at McGuire was irradiated for three reactor cycles and showed good mechanical integrity. Several full regions of VANTAGE 5 fuel are currently in operation.

The results of the safety evaluations and analyses described herein lead to the following conclusions:

1. The Westinghouse VANTAGE 5 reload fuel assemblies for the Cook Nuclear Plant Unit 2 are mechanically and hydraulically compatible with the current ANF fuel assemblies, control rods, and reactor internals interfaces. The VANTAGE 5 fuel assemblies satisfy the current design bases.

2. The VANTAGE 5 fuel assembly responses under seismic and LOCA excitations were determined using the analytical model representation of the reactor core. Analysis of the 17x17 VANTAGE 5 fuel assembly component stresses and grid impact forces due to postulated faulted condition accidents verified that the VANTAGE 5 fuel assembly design is structurally acceptable.

3. Changes in the nuclear characteristics due to the transition from ANF to VANTAGE 5 fuel will be within the range normally seen from cycle to cycle due to fuel management effects.

4. Plant operating limitations given in the Technical Specifications will be satisfied with the proposed changes noted in Section 6.0 of this report. The plant can safety operate at its current licensed power of 3411 MWt with average steam generator tube plugging levels up to 10% and peak plugging up to 15%. A reference is established upon which to base Westinghouse reload safety evaluations for future reloads with VANTAGE 5 fuel.

2.0 MECHANICAL EVALUATION

This Section evaluates the mechanical design and the compatibility of the 17x17 VANTAGE 5 fuel assembly with the current 17x17 ANF fuel assemblies during the transition through mixed-fuel cores to all VANTAGE 5 fuel cores at the Cook Nuclear Plant Unit 2. The VANTAGE 5 fuel assembly has been designed to be compatible with Westinghouse designed LOPAR and Optimized Fuel Assemblies (OFA), reactor internals interfaces, the fuel handling equipment, and refueling equipment.

The VANTAGE 5 design is compatible with and is an acceptable replacement for the Cook Nuclear Plant Unit 2 containing fuel of the ANF 17x17 design. The VANTAGE 5 design dimensions, as shown in Figure 2.1, are essential equivalent to the ANF 17x17 design from an exterior assembly envelope and reactor internals interface standpoint. Table 2.1 provides a comparison of the VANTAGE 5 and ANF 17x17 fuel assembly design parameters. The design basis and design limits for VANTAGE 5 are essentially the same as those for the Westinghouse LOPAR design.

The significant new mechanical features of the VANTAGE 5 design relative to the current ANF 17x17 fuel assembly include the following:

- Integral Fuel Burnable Absorber (IFBA)
- Intermediate Flow Mixer (IFM) Grids
- Reconstitutable Top Nozzle (RTN)
- Extended Burnup Capability
- Axial Blankets

The VANTAGE 5 fuel assembly design for Cook Nuclear Plant Unit 2 cycle 8 operation will also include the Debris Filter Bottom Nozzle (DFBN). The debris filter feature will reduce the possibility of fuel rod damage due to debris-induced fretting.

Fuel Rod Performance

Fuel rod design evaluations for the VANTAGE 5 fuel are performed using the NRC approved models in References 5 and 6 and the NRC approved extended burnup design methods in Reference 3 to demonstrate that all fuel rod design bases are satisfied.

There is no effect from a full rod design standpoint due to having fuel with more than one type of geometry simultaneously residing in the core during the transition cycles. The mechanical fuel rod design evaluation for each region incorporates all appropriate design features of the region, including any changes to the fuel rod or pellet geometry from that of previous fuel regions. Analysis of IFBA rods includes any geometry changes necessary to model the presence of the burnable absorber, and conservatively models the helium gas release from the ZrB_2 coating.

Fuel performance evaluations are completed for each fuel region to demonstrate that the design criteria will be satisfied for all fuel rod types in the core under the planned operating conditions. Any changes from the plant operating conditions originally evaluated for the mechanical design of a fuel region (for example, a power uprating or an increase in the peaking factors) are addressed for all affected fuel regions when the plant change is to be implemented.

Grid and Guide Thimble Assemblies

VANTAGE 5 top and bottom grids are fabricated from Inconel with intermediate structural grids being fabricated from Zircaloy-4. The ANF spacer grids are bi-metallic and are constructed from Zircaloy-4 with Inconel springs. The VANTAGE 5 top and bottom Inconel grids (non-mixing vane type) have a snag-resistant design feature which minimizes assembly interaction during core loading/unloading. The VANTAGE 5 Inconel grids are also similar in design to the Inconel grids of the Westinghouse LOPAR fuel assemblies. Design differences between Westinghouse VANTAGE 5 and LOPAR fuel assemblies include 1) the grid spring and dimple heights have been modified to accommodate a reduced diameter fuel rod, 2) the grid spring force has been reduced in the top grid and 3) grid straps are somewhat thicker and higher.

The Intermediate Flow Mixer (IFM) grids shown in Figure 2 are located in the three uppermost spans between the Zircaloy-4 mixing vane structural grids and incorporate a similar mixing vane array. The primary function of the IFM grid is to provide mid-span flow mixing in the hottest fuel assembly spans. Each IFM grid cell contains four dimples which are designed to prevent mid-span channel closure in the spans containing IFMs and fuel rod contact with the IFM mixing vanes. This simplified cell arrangement allows for short grid cells so that the IFM grid can accomplish its flow mixing objective with minimal pressure drop.

The IFM grids are not intended to be structural members. The outer strap configuration was designed similar to current fuel designs to preclude grid hang-up and damage during fuel handling. Additionally, the grid envelope is smaller which further minimizes the potential for damage and reduces calculated forces during seismic/LOCA events. A coolable geometry is, therefore, assured at the IFM grid elevation, as well as at the structural grid elevation.

The VANTAGE 5 guide thimble ID provides an adequate nominal diametral clearance of 0.061 inch for the control rods. For accident analyses, a 2.7 seconds scram time to the dashpot is used for the VANTAGE 5 assembly. The 0.5 second incrase in rod drop time for VANTAGE 5 as compared to LOPAR is due mainly to larger fuel assembly pressure drop attributed to VANTAGE 5 IFM grids. The increase in pressure drop results in increased RCCA resistance during rod drop excursions. Using conservative analytical techniques, the results of rod drop time calculations for Cook Nuclear Plant Unit 2 indicate that the specific scram time to the VANTAGE 5 dashpot is within the 2.7 seconds Technical Specifications limit. Thus, all safety limits associated with RCCA scram are satisfied. The VANTAGE 5 thimble tube ID provides sufficient diametral clearance for burnable absorber and source rods.

Mechanical Compatibility of Fuel Assemblies

Based on evaluations of design differences, it is concluded that VANTAGE 5 is mechanically compatible with both ANF and Westinghouse LOPAR fuel assemblies. The VANTAGE 5 fuel rod mechanical design bases remain unchanged from the Westinghouse LOPAR fuel assemblies used previously in Cook Nuclear Plant Unit 2.

Rod Bow

It is predicted that the 17x17 VANTAGE 5 fuel rod bow magnitudes will be bounded by by Westinghouse 17x17 LOPAR assembly rod bow data. The current NRC approval methodology for comparing rod bow for different assembly designs is given in Reference 7.

Rod bow in the VANTAGE 5 fuel rods containing IFBAs is not expected to differ in magnitude or frequency from that currently observed in both Westinghouse LOPAR and OFA fuel rods under similar operating conditions. No indications of abnormal rod bow have been observed during visual or dimensional inspections performed on test IFBA rods. Rod growth measurements were also within predicted bounds.

Fuel Rod Wear

Fuel rod wear is dependent on both the support provided by the fuel assembly and the flow environment to which it is subjected. Due to the VANTAGE 5 fuel assembly design employing different guide thimble tube diameter as compared to the ANF 17x17 design in addition to intermediate flow mixer (IFM) grids, an unequal axial pressure distribution results between the ANF and VANTAGE 5 fuel assembly designs. Because of the major hardware differences between ANF 17x17 and VANTAGE 5 design, evaluations were performed to evaluate hydraulically compatability of the two designs.

Hydraulic compatibility of VANTAGE 5 and ANF 17x17 fuel assembly designs was demonstrated by 1) showing that the ANF 17x17 fuel assembly was hydraulically compatible to the Westinghosue 17x17 OFA fuel assemblies, 2) referring to the study that showed Westinghouse 17x17 OFA fuel assemblies are hydraulically compatible with VANTAGE 5 fuel assemblies and 3) making direct analyses of hydraulic compatibility of the ANF 17x17 fuel assemblies to the VANTAGE 5 fuel assemblies.

The aforementioned evaluation demonstrated the ANF 17x17 fuel assembly design to be hydraulically compatible with the Westinghouse 17x17 OFA design. Evaluations have also been performed to demonstrate compatibility of the Westinghouse VANTAGE 5 and LOPAR fuel assembly designs. VANTAGE 5 fuel rod wear predictions were extrapolated from full scale hydraulic test of a VANTAGE 5 assembly adjacent to a 17x17 OFA assembly since vibration test results indicated that the crossflow effects produced by this fuel assembly combination would have the most detrimental effect on fuel rod wear.

Results of wear inspection and analysis discussed in Reference 2, Appendix A.1.4 revealed that the VANTAGE 5 fuel assembly wear characteristic was similar to that of the 17x17 OFA when both sets of data were normalized to the same test duration time. It was concluded that the VANTAGE 5 fuel rod wear would be less that the maximum wear depth established, Reference 8, for the 17x17 OFA at end-of-life condition.

In the hydraulic test of the 17x17 Optimized Fuel Assembly, some grid cell sizes were set such that small gaps existed between the grid support points and fuel rod clad. Other cells were set with

various values of spring preload. These grid/clad support conditions compare favorably with those used in the fretting wear test performed on the ANF 17x17 proof-of-fabrication fuel assembly. The clad wear results indicative of hydraulic testing of the 17x17 Optimized Fuel Assembly with gaps and minimum preload is a conservative prediction of the 17x17 ANF wear during transition.

Seismic/LOCA Impact on Fuel Assemblies

An evaluation of the VANTAGE 5 fuel assembly structural integrity considering the lateral effect of LOCA and seismic loading has been performed.

The VANTAGE 5 fuel assembly is structurally equivalent to the LOPAR and ANF fuel designs. The main differences between these designs are six Zircaloy-4 grids, three additional IFM grids, and optimized fuel rods. The load bearing capability for the Zircaloy-4 grids and flow mixers under the faulted condition loadings has been analyzed. The results indicated that 17x17 VANTAGE 5 grid loads are well below the grid strengths.

Based on the grid load results, the 17x17 VANTAGE 5 Zircaloy-4 grid is capable of maintaining the core coolable geometry under the combined Design Basis Earthquake and asymmetric pipe rupture transients in either all VANTAGE 5 or transition core operations. The 17x17 VANTAGE 5 fuel assembly is structurally acceptable for Cook Nuclear Plant Unit 2. This is also true for a transition core composed of VANTAGE 5, ANF and LOPAR fuel assembly core configurations. The grids of either fuel type will not buckle due to combined impact loads of seismic and LOCA events. There is no flow channel reduction during a LOCA; thus, the coolable geometry requirement is met. The stresses in the fuel assembly components resulting from seismic and LOCA induced deflections are well within acceptable limits.

2.8 Core Components

The core components for Cook Nuclear Plant Unit 2 are designed to be compatible with both LOPAR and VANTAGE 5 fuel assemblies. The LOPAR and VANTAGE 5 thimble tubes provide sufficient clearance for insertion of control rods and thimble plugging devices to assure proper operation of these components and fuel assembly. During Cycle 8 operation of Cook Nuclear Plant Unit 2, core components containing secondary source assemblies are restricted to locations consistent with ANF fuel assemblies.

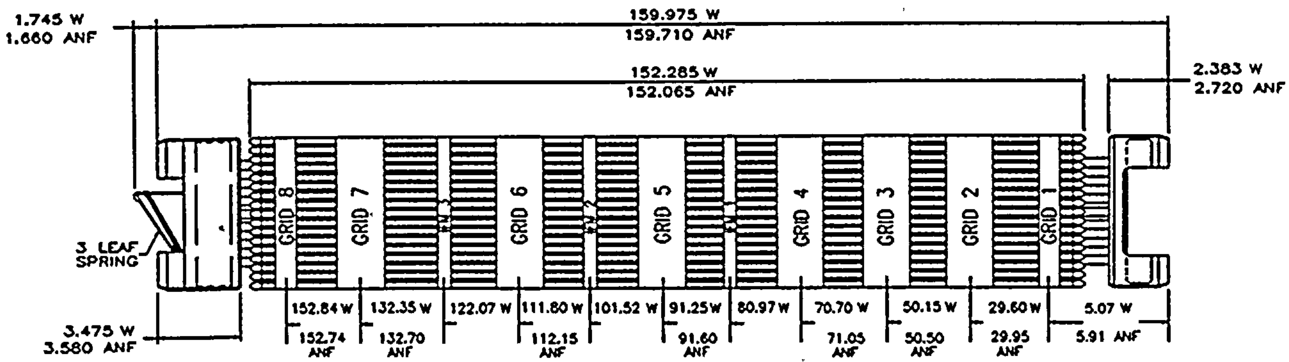
The thimble plugs included in the plugging devices for Cook Nuclear Plant Unit 2 have been designed to be compatible with both LOPAR and VANTAGE 5 designs from a mechanical and thermal/hydraulic perspective. The ANF thimble tube ID is enveloped by both LOPAR and VANTAGE 5 designs; thus, the thimble plugs are also compatible with ANF fuel assemblies.

TABLE 2.1

Comparison of ANF 17x17
and
W 17x17 VANTAGE 5 Assembly Design Parameters

PARAMETER	ANF 17x17 DESIGN	W 17x17 VANTAGE 5 DESIGN
Fuel Assy Length, in	159.710	159.975
Fuel Rod Length, in	152.065	152.285
Assembly Envelope, (width), in	8.426	8.426
Compatible with Core Internals	Yes	Yes
Fuel Rod Pitch, in	.496	.496
Number of Fuel Rods/Assy	264	264
Number/Guide Thimble Tubes/Assy	24	24
Number/Instrumentation Tube/Assy	1	1
Fuel Tube Material	Zircaloy-4	Zircaloy-4
Fuel Rod Clad OD, in	0.360	0.360
Fuel Rod Clad Thickness, in	0.0250	0.0225
Fuel/Clad Radial Gap, mil	3.5	3.1
Fuel Pellet Diameter, in	.3030	.3088
Fuel Pellet Length		
Enriched Fuel, in	.348	.370
Unenriched Fuel, in	N/A	.500
Guide Thimble Material	Zircaloy-4	Zircaloy-4
Guide Thimble OD (above dashpot), in	.480	.474

SCHEMATIC OF WESTINGHOUSE 17x17 VANTAGE 5 FUEL ASSEMBLY



W – WESTINGHOUSE 17x17 VANTAGE 5 FUEL ASSEMBLY DIMENSION+

ANF – ADVANCED NUCLEAR FUEL 17x17 FUEL ASSEMBLY DIMENSION+

ANF GRID WIDTH
- 2.50 (valley to valley)
- 2.96 (peak to peak)

WESTINGHOUSE GRID WIDTH
- top and bottom grid 1.52 *
- mid grids 2.25 *
- ifm grids .475 *

* based on inner strap width
\+ dimensions are in inches (nominal)

Cook Nuclear Plant Unit 2

FIGURE 2.1

COMPARISON OF WESTINGHOUSE VANTAGE 5 and ANF 17x17 FUEL ASSEMBLY DIMENSIONS

3.0 NUCLEAR EVALUATION

The evaluation of the transition and equilibrium cycle VANTAGE 5 cores presented in Reference 2, as well as the Cook Nuclear Plant Unit 2 specific transition core evaluations, demonstrate that the impact of implementing VANTAGE 5 does not cause a significant change to the physics characteristics of the Cook Nuclear Plant Unit 2 cores beyond the normal range of variations seen from cycle to cycle.

The nuclear design philosophy, methods and core models used in the Cook Nuclear Plant Unit 2 reload transition core evaluations are described in References 2, 4, 9, 10 and 11. These licensed methods and core models have been used for Donald C. Cook Unit 1 and other previous Westinghouse reload designs using the OFA and VANTAGE 5 fuel. No changes from the above reference to the nuclear design philosophy, methods, or core models are necessary because of the transition to VANTAGE 5 fuel.

Based on the nuclear evaluation, the following Cook Nuclear Plant Unit 2 Technical Specifications changes are proposed:

1) Increased $F^N_{\Delta H}$ limits. These higher limits will allow loading pattern designs with reduced neutron leakage which in turn will allow longer cycles.

2) Increased F_Q limit. This increased limit will provide greater flexibility with regard to accommodating the axially heterogeneous cores (axial blankets and reduced length burnable absorbers)

Power distributions and peaking factors show slight changes as a result of the incorporation of axial blankets and reduced length IFBAs in addition to the normal variations experienced with different loading patterns. The usual methods of enrichment variation and burnable absorber usage can be employed in the transition and full VANTAGE 5 cores to ensure compliance with the peaking factor Technical Specifications.

Evaluation of the key safety analysis parameters for the Cook Nuclear Plant Unit 2 reactor as it transitions to an all VANTAGE 5 core shows that the changes in values of the key safety analysis

parameters are typical of the normal cycle-to-cycle variations experienced as loading patterns change. As is current practice, each reload core design will be evaluated to assure that design and safety limits are satisfied according to the reload methodology. The design and safety limits will be documented in each cycle specific Reload Safety Evaluation (RSE) report which serves as a basis for any significant changes which may require a future NRC review.

4.0 THERMAL AND HYDRAULIC EVALUATION

The analysis of the ANF and VANTAGE 5 fuel is based on the Revised Thermal Design Procedure (RTDP) described in Reference 12. The ANF fuel analysis uses the W-3 DNB correlation described in References 13 and 14 and the VANTAGE 5 fuel uses the WRB-2 DNB correlation described in Reference 2. A 0.88 multiplier is applied to the W-3 DNB correlation to account for the 17x17 fuel rod diameter effect. The WRB-2 DNB correlation takes credit for the VANTAGE 5 fuel assembly mixing vane design. In addition the W-3 DNB correlation is used where appropriate. Table 4.1 summarizes the pertinent thermal and hydraulic design parameters.

The design method employed to meet the DNB design basis is the RTDP which has been approved by the NRC, Reference 12. Uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters and computer codes are statistically combined with the DNB correlation uncertainties such that there is at least a 95 percent probability at a 95% confidence level that DNB will not occur on the most limiting fuel rod during normal operation and operational transients and during transient conditions arising from faults of moderate frequency (Condition I and II events as defined in ANSI N18.2). This gives the design limit DNBRs. Since the parameter uncertainties are considered in determining the design limit DNBR values, then the plant safety analyses are performed using values of input parameters without uncertainties. The design limit DNBRs are 1.23 and 1.22 for the typical and thimble cells respectively for VANTAGE 5 fuel and 1.39 and 1.36 for the typical and thimble cells respectively for ANF fuel. Standard Thermal Design Procedure (STDP) is used where the RTDP methodology is not applicable. In the STDP method the parameters used in analysis are treated in a conservative way so as to give the lowest minimum DNBR.

In addition to the above considerations, a plant specific DNBR margin has been considered in the analysis. In particular, DNBR safety analysis limits of 1.43 and 1.40 for the typical and thimble cells respectively for ANF fuel, and 1.69 and 1.61 for the typical and thimble cells respectively for VANTAGE 5 fuel were employed in the safety analyses. The differences between the design and safety analysis limits result in DNBR margin. A fraction of the margin is utilized to accommodate the transition core penalty. For VANTAGE 5 fuel this transition core penalty is a function of the number of VANTAGE 5 fuel assemblies in the core as given in Reference 15 and is based on a maximum value of 12.5%. There is no transition core penalty for ANF fuel for analyses using

cosine or positive axial offset axial power shapes. The transition core penalties for ANF fuel that occur with power shapes having large negative axial offsets are accounted for in the specific analyses that use these shapes. Additional margin is used to counteract rod bow. The fuel rod bow DNBR penalty is equal to 1.3% for VANTAGE 5 fuel (Reference 7) in the 20 inch grid spans. No rod bow penalty is required in the 10 inch grid spans. There is no rod bow penalty for ANF fuel (Reference 16). The remaining margin, after consideration of these penalties, is reserved for flexibility in the design. The plant specific DNBR margin, discussed above for RTDP, is preserved whenever STDP is used.

Hydraulic compatibility tests were performed by Combustion Engineering for the ANF 17x17 proof of fab fuel assembly. The results of these tests were compared to hydraulic test data for the VANTAGE 5 fuel assembly (Reference 2). The data show that the ANF 17x17 fuel assemblies and the VANTAGE 5 fuel assembies are hydraulically compatible.

The Westinghouse transition core DNB methodology is given in References 1 and 17 and has been approved by the NRC via Reference 18. Using this methodology, transition cores are analyzed as if they were full cores of one assembly type (full ANF or full VANTAGE 5), applying the applicable transition core penalties.

The fuel temperatures used in safety analysis calculations for the VANTAGE 5 fuel were calculated with the PAD performance code (Reference 6). This code was used to perform both design and licensing calculations. These fuel temperatures were used as initial conditions for LOCA and non-LOCA transients.

TABLE 4.1

THERMAL AND HYDRAULIC DESIGN PARAMETERS

Thermal and Hydraulic Design Parameters (Using RTDP)		Bounding Parameters for Mixed Cores (Cycles 8 & 9)	Bounding Parameters for Homogeneous VANTAGE 5 Cores (Cycle 10 & Beyond)
Reactor Core Heat Output, MWt		3588	3588
Reactor Core Heat Output, 10^6 BTU/Hr		12243	12243
Heat Generated in Fuel, %		97.4	97.4
Core Pressure, Nominal, psia		2280	2130
$F^N_{\Delta H}$ Nuclear Enthalpy Rise	(ANF)	1.49[1+.2(1-P)]*	
Hot Channel Factor	(V-5)	1.59[1+.3(1-P)]*	1.59[1+.3(1-P)]*
Safety Analysis Limit DNBR			
Typical Flow Channel	(ANF)	1.43	
	(V-5)	1.69	1.69
Thimble (Cold Wall) Flow Channel	(ANF)	1.40	
	(V-5)	1.61	1.61
DNB Correlation	(ANF)	W-3	
	(V-5)	WRB-2	WRB-2

* The 4% radial power uncertainty has been removed for statistical combination with other uncertainties in the RTDP analysis.

TABLE 4.1 (cont)

THERMAL AND HYDRAULIC DESIGN PARAMETERS

HFP Nominal Coolant Conditions		Bounding Parameters for Mixed Cores (Cycles 8 & 9)	Bounding Parameters for Homogeneous VANTAGE 5 Cores (Cycle 10 & Beyond)
Vessel Minimum Measured Flow Rate (including Bypass)			
10^6 lbm/hr		139.1	137.8
GPM		366,400	366,400
Vessel Thermal Design Flow Rate (including Bypass)			
10^6 lbm/hr		134.6	133.2
GPM		354,000	354,000
Core Flow Rate (excluding Bypass, based on Thermal Design Flow)			
10^6 lbm/hr		127.7	126.4
GPM		335,900	335,900
Fuel Assembly Flow Area for Heat Transfer, ft^2	(ANF)	53.98	
	(V-5)	54.10	54.10
Core Inlet Mass Velocity, 10^6 lbm/hr-ft (Based on TDF)	(ANF)	2.366	
	(V-5)	2.359	2.336

TABLE 4.1 (cont)

THERMAL AND HYDRAULIC DESIGN PARAMETERS

Thermal and Hydraulic Design Parameters (Based on Thermal Design Flow)		Bounding Parameters for Mixed Cores (Cycles 8 & 9)	Bounding Parameters for Homogeneous VANTAGE 5 Cores (Cycle 10 & Beyond)
Nominal Vessel/Core Inlet Temperature, ^{o}F		541.8	547.6
Vessel Average Temperature, ^{o}F		576.0	581.3
Core Average Temperature, ^{o}F		579.5	584.9
Vessel Outlet Temperature, ^{o}F		610.2	615.0
Average Temperature Rise in Vessel, ^{o}F		68.4	67.4
Average Temperature Rise in Core, ^{o}F		71.7	70.6
Heat Transfer			
Active Heat Transfer Surface Area*, ft^2	(ANF/V-5)	57,505	57,505
Average Heat Flux, $BTU/hr\text{-}ft^2$	(ANF/V-5)	207,410	207,410
Average Linear Power, kw/ft		5.72	5.72
Peak Linear Power for Normal Operation, kw/ft**		13.3	13.3

* Assumes all ANF or VANTAGE 5 core

** Based on 2.32 F_Q peaking factor

5.0 ACCIDENT EVALUATION

5.1 Non-LOCA

5.1.1 Introduction

This section addresses the impact of the complete transition of Cook Nuclear Plant Unit 2 from ANF 17x17 fuel to Westinghouse 17x17 VANTAGE 5 fuel on the FSAR Chapter 14 Non-LOCA Accident Analyses. The methods used for accident evaluation are described in Reference 4 and are discussed in further detail in Section 5.1.4.

The Cook Nuclear Plant Unit 2 licensing basis, as reported in the original FSAR (Reference 19) includes analyses or evaluations of fifteen (15) Non-LOCA accidents. These accidents are:

a. Uncontrolled RCCA Bank Withdrawal From a Subcritical Condition
b. Uncontrolled RCCA Bank Withdrawal at Power
c. Rod Cluster Control Assembly (RCCA) Misalignment
d. Rod Cluster Control Assembly (RCCA) Drop
e. Uncontrolled Boron Dilution
f. Loss of Forced Reactor Coolant Flow
g. Startup of an Inactive Reactor Coolant Loop
h. Loss of External Electric Load or Turbine Trip
i. Loss of Normal Feedwater
j. Excessive Heat Removal due to Feedwater System Malfunction
k. Excessive Load Increase
l. Loss of Offsite Power (LOOP) to the Station Auxiliaries
m. Rupture of a Steamline (Steamline Break)
n. Rupture of a Control Rod Drive Mechanism (CRDM) Housing (Rod Cluster Control Assembly Ejection)
o. Major Rupture of Main Feedwater Pipe (Feedline Break)

All of the above fifteen Non-LOCA accidents have been reviewed to address any impact resulting from the VANTAGE 5 fuel reload. The specific design associated with the VANTAGE 5 fuel and the modified safety analysis assumptions that were considered in the Non-LOCA safety analysis are described in the following sections.

5.1.2 VANTAGE 5 Design Features

The design features of this VANTAGE 5 fuel reload transition that were considered in the Non-LOCA analysis and evaluations are:

- Intermediate Flow Mixer (IFM) Grids
- Axial Blankets
- Integral Fuel Burnable Absorbers (IFBAs)
- Debris Filter Bottom Nozzle
- Reconstitutable Top Nozzle

A brief description of each of these and its consideration in the Non-LOCA safety analyses follows:

<u>IFM Grids</u>

The IFM grid feature of the VANTAGE 5 fuel design increases DNB margin. The fuel safety analysis limit DNB margin was set to ensure that the core thermal safety limits for the VANTAGE 5 fuel with an $F^N_{\Delta H}$ of 1.65 are acceptable. However, for the transition cycles the ANF fuel core thermal safety limits with $F^N_{\Delta H}$ of 1.55 are more restrictive. Thus, the more restrictive core limits correspond to the ANF fuel design. Any transition core penalty is accounted for with the available DNB margin.

The IFM grid feature of the VANTAGE 5 fuel design increases the core pressure drop. One result is that the control rod scram time to the dashpot has been increased to 2.7 seconds. This increased drop time primarily affects the fast reactivity transients which were reanalyzed for this report. The revised control rod drop time was incorporated in all the reanalyzed events requiring this parameter change. The Startup of an Inactive Reactor Coolant Loop transient not analyzed for this report has been evaluated for this parameter change.

Axial Blankets and IFBAs

Axial blankets reduce power at the ends of the rod which increases axial peaking at the interior of the rod. This effect is offset by the presence of part length IFBAs which flatten the power distribution. The net effect on the axial shape is a function of the number and configuration of IFBAs in the core and the time in core life. The effects of axial blankets and IFBAs on the reload safety analysis parameters are taken into account in the reload design process. The axial power distribution assumption in the safety analyses kinetics calculations have been determined to be sufficiently conservative to accommodate the introduction of axial blankets in the Cook Nuclear Plant Unit 2. Figure 5.1 shows the axial power distribution assumed in the Non-LOCA safety analyses.

Reconstitutable Top Nozzle (RTN) and Debris Filter Bottom Nozzle (DFBN)

Reconstitutable Top Nozzles (RTN) and Debris Filter Bottom Nozzles (DFBN) have been used extensively in Westinghouse designs. Analysis was performed to confirm the hydraulic compatibility of the Westinghouse nozzle designs to the existing ANF designs and therefore, will not impact any parameters important to the Non-LOCA safety analyses.

5.1.3 Modified Safety Analysis Assumptions

Listed below are the analysis assumptions which represent a departure from that currently used for Cook Nuclear Plant Unit 2.

- Revised Maximum Moderator Density Coefficient
- Increased Design Enthalpy Rise Hot Channel Factors ($F^N_{\Delta H}$) and F_Q for the Westinghouse VANTAGE 5 fuel
- Increase $F^N_{\Delta H}$ Part Power Multiplier on Westinghouse VANTAGE 5 fuel
- Decreased Shutdown Margin
- Revised Thermal Design Procedure (RTDP)
- Increased Core Power
- Reduced Temperature and Pressure (RTP) Operation
- 0 ppm boron concentration in the Boron Injection Tank (BIT)
- Constant Steam Generator Level Program
- System Performance Degradation

A brief description of each of these assumptions follows:

Revised Maximum Moderator Density Coefficient

The analyses consider an End-of-Cycle (EOC) Life most positive Moderator Density Coefficient (MDC) of 0.54 Δk/gm/cc. The Moderator Temperature Coefficient (MTC) as a function of vessel average temperature is shown in Figure 5.2.

Increased $F^N_{\Delta H}$ and F_Q

The design $F^N_{\Delta H}$ for the ANF and VANTAGE 5 fuel is 1.55 and 1.65 respectively. The Non-LOCA calculations applicable for the VANTAGE 5 core have assumed a full power $F^N_{\Delta H}$ of 1.65. This is a conservative safety analysis assumption for this report.

The increase in the Technical Specification maximum LOCA F_Q from 2.1 to 2.22 is conservatively bounded in the Non-LOCA transients. A maximum F_Q of 2.5 was assumed in the Non-LOCA safety analyses.

Increased $F^N_{\Delta H}$ Part Power Multipliers

The $F^N_{\Delta H}$ part power multipliers are 0.2 for ANF fuel and 0.3 for VANTAGE 5 fuel. These values have been considered in the generation of the core thermal limits for both fuel types. The changes in the core thermal safety limits result in a change to the Overtemperature and Overpower ΔT (OTΔT/OPΔT) reactor protection trip setpoints. Two sets of OTΔT/OPΔT setpoints were calculated. The first set of these setpoints is calculated based on ANF core thermal limits and is applicable for transition cycles. The second set of these setpoints is calculated based on VANTAGE 5 core thermal limits and is applicable for full VANTAGE 5 core (Cycles 10 and beyond). DNB analyses which are performed using LOFTRAN (see Appendix B, Reference 5) alone were analyzed twice, once for mixed core cycles and once for full VANTAGE 5 core. The remaining DNB analyses have accounted for the variation in $F^N_{\Delta H}$ part power multipliers between a mixed core and a full VANTAGE 5 core.

Decreased Shutdown Margin (SDM)

A change in the shutdown margin from 2.0% Δk/k to 1.3% Δk/k was considered in the Non-LOCA safety analyses.

Revised Thermal Design Procedure (RTDP)

The calculational method utilized to meet the DNB design basis is the RTDP, which is discussed in Reference 12. Uncertainties in the plant operating parameters are statistically treated such that there is at least a 95 percent probability at a 95 percent confidence level that the minimum DNBR will be greater than the applicable limits as discussed in Section 4.2. Since the parameter uncertainties are considered in determining the design DNBR value, the plant safety analyses are performed using nominal initial conditions without uncertainties. The ANF fuel analyses used the W-3 correlation, while the VANTAGE 5 fuel analyses use the WRB-2 correlation.

Increased Core Thermal Power

An increase in the nominal core thermal power from 3411 MWt to 3588 MWt was considered in the Non-LOCA safety analyses for the potential rerating of the Cook Nuclear Plant Unit 2. The Non-LOCA safety analyses performed at 3588 MWt will conservatively bound the current nominal core thermal power level of 3411 MWt.

Reduced Temperature and Pressure (RTP) Operation

Reduced temperature and pressure operation for Cook Nuclear Plant Unit 2 was considered in the Non-LOCA safety analyses. The full power vessel average temperature range of 547 °F to 581.3 °F at either of two values of pressurizer pressure (2100 psia or 2250 psia) was considered. However, because of the DNB constraints associated with the presence of ANF fuel during transition cycles (Cycles 8 and 9), a limitation on pressure and temperature conditions will apply. These include a full power vessel average temperature range of 547 °F to 576 °F, and a pressurizer pressure of 2250 psia (see Table B.2-1, cases 2 and 3 in Appendix B). Generating an acceptable nominal setpoint for the OTΔT reactor trip setpoint during transition cycles has resulted in this limitation. This limitation will not apply when a full VANTAGE 5 core is in place. The Non-LOCA safety analyses presented in this report provide support for a "full window" (see Appendix B Table B.2-1, cases 4-7) of operation in the assumed range of RTDP operation when a full VANTAGE 5 core is in place at Cook Nuclear Plant Unit 2.

BIT Boron Concentration

A zero (0) ppm BIT boron concentration was assumed in the Non-LOCA analyses to support BIT removal at Cook Nuclear Plant Unit 2. This is a conservative safety analysis assumption for this report.

Steam Generator Water Level Program

A change in the steam generator water level program was considered in the Non-LOCA safety analyses. The existing steam generator water level program is a ramp function from 33% narrow range span (NRS) to 44% NRS from 0% power to 20% power and a constant level at 44% NRS between 20% power and 100% power. The steam generator water level program to be implemented at the beginning of Cycle 8 is a constant level at 44% NRS between 0% power and 100% power.

System Performance Degradation

The system performance degradation assumptions made for the Non-LOCA safety analyses are as follows:

- A 10% average steam generator tube plugging level. This is a conservative safety analysis assumption for the Non-LOCA analyses presented in this report and bounds a 0% tube plugging level.

- An increase in the Main Steamline Isolation Valve (MSIV) closure time from 5 seconds to 8 seconds with a corresponding increase in total response times.

- A 10% Safety Injection Flow degradation.

- A minimum required auxiliary feedwater flow rate of 450 gpm corresponding to the steam generator safety valve set pressure of 1123 psia was assumed for the Loss of Normal Feedwater analysis. For Loss of Offsite power to the Station Auxiliaries, a minimum auxiliary feedwater flow of 430 gpm corresponding to the steam generator safety valve set pressure of 1133 psia was assumed. A minimum auxiliary feedwater flow of 600 gpm corresponding to the steam generator safety valve set pressure of 1133 psia was assumed for the Feedline Break analysis.

5.1.4 Non-LOCA Safety Evaluation Methodology

The Non-LOCA safety evaluation process is described in Reference 4. The methodology determines if a core configuration is bounded by existing safety analyses in order to confirm that applicable safety criteria are satisfied. The methodology systematically identifies parameter changes

on a cycle-by-cycle basis which may exceed existing safety analysis assumptions and identifies the transients which require reevaluation. This methodology is applicable to the evaluation of VANTAGE 5 transition and full cores.

Any required reevaluation identified by the reload methodology is one of two types. If the identified parameters is only slightly out of bounds, or the transient is relatively insensitive to that parameter, a simple evaluation may be made which conservatively evaluates the magnitude of the effect and explains why the actual analysis of the event does not have to be repeated. Alternatively, should the deviation be large and/or expected to have a significantly or not easily quantifiable effect on the transients, reanalyses are required.

The reanalysis approach will utilize Westinghouse codes and methods which have been accepted by the NRC and have been used in previous submittals to the NRC. These methods are those which have been presented to the NRC for a specific plant, reference SARs or reports for NRC approval. The analysis methods and codes are described in Appendix B.

With the exception of the Startup of an Inactive Loop, all the Non-LOCA accidents listed in Section 5.1:1 have been reanalyzed for this report. In accordance with the Technical Specification 3/4.4.1 (Amendment 59), Cook Nuclear Plant Unit 2 operation during Modes 1 and 2 with less than four loops is not permitted. Since three loop operation during Modes 1 and 2 is prohibited, the Startup of an Inactive Reactor Coolant Loop event was not considered for the transition to VANTAGE 5 fuel.

The key safety parameters are documented in Reference 4. Values of these safety parameters which bound both fuel types (ANF and VANTAGE 5) were assumed in the Non-LOCA safety analyses. For subsequent fuel reloads, the key safety parameters will be evaluated to determine if violations of these bounding values exist. Reevaluation of the affected accidents will take place as described in Reference 4.

5.1.5 Conclusions

Descriptions of the Non-LOCA accidents reanalyzed for this report, method of analysis, results, and conclusions are contained in Appendix B. Based on the plant operating limitations given in the Technical Specifications and the proposed Technical Specifications changes given in Section 6.0 of

this report, the results show that the transition from ANF to 17x17 VANTAGE 5 fuel, including the aforementioned modified safety analysis assumptions described in Section 5.1.3, can be accommodated with margin to the applicable UFSAR safety limits.

The impact of the transition to VANTAGE 5 fuel on Steam Line Break Mass and Energy Releases for both inside and outside containment is addressed in Section 5.4.

5.2 LOCA

5.2.1 Large Break LOCA

5.2.1.1 Description of Analysis/Assumptions for 17X17 VANTAGE 5 Fuel

The large break Loss-Of-Coolant Accident (LOCA) analysis for Cook Nuclear Plant Unit 2, applicable to a full core of VANTAGE 5 fuel assemblies, was performed to develop Cook specific peaking factor limits. This is consistent with the methodology employed in the Reference Core Report for 17X17 VANTAGE 5, Reference 2. The Westinghouse 1981 Evaluation Model with BASH, References 20 and 21, was utilized and a spectrum of cold leg breaks were analyzed for Cook Nuclear Plant Unit 2 that bounds high and low pressure and high and low temperature operation. Other pertinent analysis assumptions include: a core thermal power of 3588 MWt, 15% steam generator tubes plugged in each of four steam generators (i.e. uniform among the loops); an F_Q of 2.22, an $F^N_{\Delta H}$ of 1.62, and fuel data based on the new fuel thermal model, Reference 6. The most limiting break determined from the high temperature/high pressure analysis was reanalyzed at the reduced temperature and reduced pressure conditions.In addition a case was analyzed to consider the closure on the RHR crosstie valves. This case was at 3413 MWt with the 95% part-power values of 2.335 and 1.644 for F_Q and $F^N_{\Delta H}$ respectively. The analysis assumptions, results, tables and figures are presented in Appendix C.

Section 2.0, Mechanical Design, demonstrates that the ANF 17x17 fuel assemblies currently in operation in Cook Nuclear Plant Unit 2 are very similar to the Westinghouse 17x17 VANTAGE 5 5 fuel assemblies in terms of geometric characteristics. Section 4.3 demonstrates that the 17x17 ANF fuel assembly is nearly identical to the Westinghouse 17x17 OFA assembly in terms of hydraulic characteristics. Therefore, the analyses reported in Reference 2 which demonstrate that the 17x17 VANTAGE 5 fuel features result in a fuel assembly that is more limiting than a Westinghouse 17x17 OFA fuel assembly, with respect to large break LOCA ECCS performance, remain valid as applied at Cook Nuclear Plant Unit 2. The same large break LOCA transition core

penalty reported in Section 5.2.3 of Reference 2 will be applied to the transition from 17x17 ANF fuel assemblies to Westinghouse 17x17 VANTAGE 5 fuel assemblies.

In addition, those ANF assemblies which remain in the core during transition to a full core of Westinghouse 17x17 VANTAGE 5 fuel have lower F_Q and $F^N_{\Delta H}$ limits (as specified in the Core Operating Limits Report). This provides additional assurance that the computed Peak Clad Temperature (PCT) for an entire core of Westinghouse 17x17 VANTAGE 5 fuel assemblies, including an appropriate transition core penalty, constitute a bounding analysis for the Cook Nuclear Plant Unit 2. As such, VANTAGE 5 fuel has been analyzed herein.

5.2.1.2 Method of Analysis

The methods used to analyze the large break LOCA accident for Cook Nuclear Plant Unit 2 for VANTAGE 5 fuel, including computer codes used and assumptions are described in detail in Appendix C, Section C.3.1.2.

5.2.1.3 Results

The results of this analysis, including tabular and plotted results of the break spectrum analyzed are provided in Appendix C, Section C.3.1.2, which has been prepared using the NRC Standard Format and Content Guide, Regulatory Guide 1.70, Revision 2 for accidents applicable to Cook Nuclear Plant Unit 2.

Reference 20 states three restrictions related to the use of the 1981 Evaluation Model (EM) with BASH calculational model. The application of these restrictions to the plant specific large break LOCA analysis was addressed with the following conclusions:

Cook Nuclear Plant Unit 2 is neither an Upper Head Injection (UHI) nor Upper Plenum Injection (UPI) plant so restriction 1 does not apply.

The Cook Nuclear Plant Unit 2 plant specific LOCA analysis analyzed both minimum and maximum ECCS cases to address restriction 2. The C_D = 0.6 Double Ended Cold Leg Guillotine (DECLG) break with minimum ECCS flows was found to result in the most limiting consequences.

Generic sensitivity studies were performed by Westinghouse for a typical 4-loop plant using different power shapes. This sensitivity study demonstrated that the chopped cosine was the most limiting power shape, Reference 21. A chopped cosine power shape was used in the large break LOCA analysis for Cook Nuclear Plant Unit 2, thus satisfying restriction 3.

5.2.1.4 Conclusions

The large break LOCA analysis performed for the Cook Nuclear Plant Unit 2 has demonstrated that for breaks up to a double-ended severance of the reactor coolant piping, the Emergency Core Cooling System (ECCS) will meet the acceptance criteria of Title 10 CFR Part 50 Section 46. That is:

1. The calculated peak cladding temperature will remain below the required 2200 ^{o}F.

2. The amount of fuel cladding that reacts chemically with the water or steam to generate hydrogen does not exceed 1% of the hypothetical amount that would be generated if all the zirconium metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

3. The localized cladding oxidation limit of 17 percent is not exceeded during or after quenching.

4. The core remains amenable to cooling during and after the LOCA.

5. The core temperature is reduced and decay heat is removed for an extended period of time. This is required to remove the heat produced by the long-lived radioactivity remaining in the core.

The time sequence of events for all breaks analyzed is shown in Table C.3.1-5 of Appendix C, Section C.3.1.2.

The large break LOCA analysis for Cook Nuclear Plant Unit 2 assuming a full core of VANTAGE 5 fuel; utilizing the 1981 EM with BASH calculational model, resulted in a peak cladding temperature of 2140_{o}F for the limiting $C_D = 0.6$ DECLG break at a total peaking factor

of 2.22. The maximum local metal-water reaction was 6.80% and the total core wide metal-water reaction was less than 0.3% for all cases analyzed. Further, the clad temperature transients reached a maximum at a time when the core geometry was still amenable to cooling.

The effect of the transition core cycles is conservatively evaluated to be at most 50 °F higher in calculated peak cladding temperature which would yield a transition core PCT of 2190 °F. The transition core penalty can be accommodated by the margin to the 10 CFR 50.46, 2200 °F limit. It can be determined from the results contained in Appendix C, Section C.3.1.2 that the large break LOCA ECCS analysis for the Cook Nuclear Plant Unit 2 remains in compliance with the requirements of 10CFR50.46 including consideration for transition core configurations.

5.2.2 Small Break LOCA

5.2.2.1 Description of Analysis and Assumptions for 17X17 VANTAGE 5

Consistent with the logic presented in Section 5.2.1.1 for large break LOCAs, the small break loss-of-coolant accident (LOCA) was analyzed assuming a full core of VANTAGE 5 fuel to determine the peak cladding temperature. As with the large break LOCA, the methodology employed in WCAP-10444-P-A, Reference 2, for transitioning from Westinghouse 17x17 OFA to 17x17 VANTAGE 5 fuel was applied to the transition from 17x17 ANF fuel assemblies to Westinghouse 17x17 VANTAGE 5 fuel assemblies. The currently approved NOTRUMP Small Break ECCS Evaluation Model, Reference 22, was utilized for a spectrum of cold leg breaks. Appendix C, Section C.3.2, includes a full description of the analysis and assumptions utilized for the Westinghouse VANTAGE 5 ECCS Small Break LOCA analysis. Pertinent assumptions include an $F^N_{\Delta H}$ of 1.62 for a full core of 17x17 VANTAGE 5 fuel assemblies in the Cook Nuclear Plant Unit 2 core, a total peaking factor corresponding to 2.32 at the core mid-plane, 15% steam generator tube plugging, and a core thermal power level of 3588 MWt. The most limiting small break LOCA was computed for the low pressure/high temperature case and the limiting break size was reanalyzed for two additional cases to cover the range of operating temperatures and pressures being considered. An additional small break LOCA calculation was made which assumed that the HHSI cross tie valves were closed. To compensate for the reduction in safety injection due to closure of the cross tie valves, reactor power was reduced to 3413 MWt.

Sensitivity studies performed using the NOTRUMP small break evaluation model have demonstrated that VANTAGE 5 fuel is more limiting than OFA fuel in calculated ECCS

performance. It has been previously demonstrated that the 17x17 ANF fuel assemblies are essentially identical in both geometry and hydraulic characteristics to the Westinghouse 17x17 OFA fuel assembly. Therefore, the conclusion of Reference 2 that a small break LOCA analysis for a full core of Westinghouse 17x17 VANTAGE 5 fuel is bounding, remains valid. On this basis, only VANTAGE 5 fuel was analyzed, since it is the most limiting of the two types of fuel (17x17 ANF and Westinghouse 17x17 VANTAGE-5) that would reside in the core for Cook Nuclear Plant Unit 2.

5.2.2.2 Method of Analysis

The methods of analysis, including codes used and assumptions, are described in detail in Appendix C, Section C.3.2.

5.2.2.3 Results

The results of this analysis, including tabular and plotted results of the break spectrum analyzed, are provided in Appendix C, Section C.3.2.

5.2.2.4 Conclusions

The small break VANTAGE 5 LOCA analysis for Cook Nuclear Plant Unit 2, utilizing the currently approved NOTRUMP Evaluation Model resulted in a peak cladding temperature (PCT) of 1357 °F for the 4-inch diameter cold leg break at high temperature and low pressure. The 4-inch break size was then used for both a low temperature/high pressure and high temperature/high pressure analysis which resulted in PCTs of 1315 °F and 1325 °F respectively. The analysis assumed a limiting small break power shape consistent with a $F_Q(Z)$ envelope of 2.32 at the core midplane elevation and 2.15 at the top of the core. The maximum local metal-water reaction is 0.15%, and the total core metal-water reaction is less than 0.3 percent for all cases analyzed corresponding to less than 0.3 percent hydrogen generation. The clad temperature transients turn around at a time when the core geometry is still amenable to cooling.

Analyses presented in Appendix C, Section C.3.2 show that one high head charging pump and one safety injection pump, together with the accumulators, provide sufficient core flooding to keep the calculated peak clad temperature well below the required limit of 10 CFR 50.46 for the Cook Nuclear Plant Unit 2. It can also be seen that the ECCS analysis remains in compliance with all other requirements of 10 CFR 50.46 and the peak cladding temperature results are below the peak

cladding temperatures calculated for the large break LOCA. Adequate protection is therefore afforded by the ECCS in the event of a small break LOCA.

5.2.3 Transition Core Effects on LOCA

When assessing the effect of transition cores on the large break LOCA analysis, it must be determined whether the transition core can have a greater calculated peak cladding temperature (PCT) than either a complete core of the 17x17 ANF assembly design or a complete core of the Westinghouse 17x17 VANTAGE 5 design. For a given peaking factor, the only mechanism available to cause a transition core to have a greater calculated PCT than a full core of either fuel is the possibility of flow redistribution due to fuel assembly hydraulic resistance mismatch. Hydraulic resistance mismatch will exist only for a transition core and is the only unique difference between a complete core of either fuel type and the transition core.

In addition, all the various LOCA related analyses discussed below have been analyzed or evaluated to include a control rod drop time of 2.7 seconds, as is required for the 17x17 VANTAGE 5 fuel.

5.2.3.1 Large Break LOCA

The large break LOCA analysis was performed with a full core of VANTAGE 5 fuel and conservatively applies the blowdown transient results to transition cores. The VANTAGE 5 differs hydraulically from the 17x17 ANF assembly design it replaces. The difference in the total assembly hydraulic resistance between the two designs is approximately 10% higher for VANTAGE 5.

An evaluation of hydraulic mismatch of approximately 10% showed an insignificant effect on blowdown cooling during a LOCA. The SATAN-VI computer code models the crossflows between the average core flow channel (average of 192 fuel assemblies) and the hot assembly flow channel (one fuel assembly) during blowdown. To better understand the transition core large break LOCA blowdown transient phenomena, conservative blowdown fuel clad heatup calculations have been performed to determine the clad temperature effect on the new fuel design for mixed core configurations. The effect was determined by reducing the axial flow in the hot assembly at the appropriate elevations to simulate the effects of the transition core hydraulic resistance mismatch. In addition, the Westinghouse blowdown evaluation model was modified to account for grid heat transfer enhancement during blowdown for this evaluation. The results of this evaluation have shown that no peak cladding temperature penalty is observed during blowdown for the mixed core.

Therefore, it is not necessary to perform a blowdown calculation for the VANTAGE 5 transition core configuration because the evaluation model blowdown calculation performed for the full VANTAGE 5 core is conservative and bounding.

Since the overall resistance of the two types of fuel is essentially the same, only the crossflows during core reflood due to Intermediate Flow Mixing grids need be evaluated. The LOCA analysis uses the BASH computer code to calculate the reflood transient, Reference 20, which utilizes the BART code, Reference 23. A detailed description of the BASH code is given in Appendix C. Fuel assembly design specific analyses have been performed with a version of the BART computer code, which accurately models mixed core configurations during reflood. Westinghouse transition core designs, including specific 17X17 OFA to VANTAGE 5 transition core cases, were analyzed. For this case, BART modeled both fuel assembly types and predicted the reduction in axial flow rates at the appropriate elevations. As expected, the increase in hydraulic resistance for the VANTAGE 5 assembly was shown to produce a reduction in reflood steam flow rate for the VANTAGE 5 fuel at mixing vane grid elevations for transition core configurations. This reduction in steam flow rate is partially offset by the fuel grid heat transfer enhancement predicted by the BART code during reflood. The various fuel assembly specific transition core analyses performed resulted in peak cladding temperature increases of up to 50 °F for core axial elevations that bound the location of the PCT. Therefore, the maximum PCT penalty possible for VANTAGE 5 fuel residing in a transition core is 50 °F, Reference 2. As stated earlier, this transition core penalty continues to apply to the transition from 17x17 ANF fuel assemblies to Westinghouse 17x17 VANTAGE 5 fuel assemblies due to the near identical design of 17x17 ANF and Westinghouse 17x17 OFA fuel assemblies. Once a full core of VANTAGE 5 fuel is achieved the large break LOCA analysis will apply without the transition core penalty.

5.2.3.2 Small Break LOCA

The NOTRUMP computer code, Reference 24, is used to model the core hydraulics during a small break LOCA event. Only one core flow channel is modeled in the NOTRUMP computer code, Reference 22, since the core flow rate during a small break LOCA is relatively slow, providing enough time to maintain flow equilibrium between fuel assemblies (i.e., no crossflow). Therefore, hydraulic resistance mismatch is not a factor for small break LOCA. Thus, it is not necessary to perform a small break LOCA evaluation for transition cores, and it is sufficient to

reference the small break LOCA for the complete core of the VANTAGE 5 fuel design, as bounding for all transition cycles.

5.2.4 Post-LOCA Long-Term Core Cooling - ECCS flows, Core Subcriticality and Switchover of the ECCS to Hot Leg Recirculation

The implementation of VANTAGE 5 fuel at the Cook Nuclear Plant Unit 2 does not affect the assumptions for decay heat, core reactivity or boron concentration for sources of water residing in the containment sump Post-LOCA. Thus, these licensing requirements associated with LOCA are not significantly affected by the implementation of VANTAGE 5 fuel.

Additionally, Westinghouse and American Electric Power Service Corp. perform an independent check on core subcriticality for each fuel cycle operated at Cook Nuclear Plant Unit 2.

5.2.5 Short-Term Containment Analysis

The containment building subcompartments are the fully or partially enclosed volumes within the containment which contain high energy lines. These subcompartments are designed to limit the adverse effects of a postulated high energy pipe rupture within them. The short term containment integrity analysis is used to verify the adequacy of interior structures and walls by demonstrating that calculated differential pressures are less than design limits. The functioning of the ice condenser is demonstrated and containment integrity is also verified. The short-term containment integrity analysis is described in Section 14.3.4.3 of the Cook Nuclear Plant Unit 2 UFSAR.

The short-term containment analysis was recently performed to support operation of the Cook Nuclear Plant Unit 2 at an uprated NSSS power level of 3600 MWt, RCS average vessel temperatures over the range of 547 °F to 581.3 °F, and at RCS pressures of 2100 psia or 2250 psia. This analysis is documented in Section 3.4.1 of WCAP-11902, Reference 25. Since the peak subcompartment pressures occur within a couple of seconds of transient initiation, the changes resulting from the VANTAGE 5 fuel reload do not affect the short-term containment analysis.

5.2.6 LOCA Containment Integrity

The long term peak containment pressure calculation was recently performed to support operation of the Cook Nuclear Plant Unit 2 with the RHR crosstie valves closed at an NSSS power level of 3425 MWt. This analysis is documented in WCAP-11908, Reference 26. The analysis documented

in WCAP-11908 also provides justification for operation at 3425 MWt NSSS power, RCS vessel average temperatures of 547 °F to 578.7 °F and RCS pressurizers pressures of 2100 psia or 2250 psia. The analysis also considers and provides justification for operation with 10% average (15% peak) steam generator tube plugging; 10% high head charging, safety injection, and residual heat removal pump degradation; initial accumulator volume of 946 ± 25 cubic feet; 10% containment spray flow rate degradation; and spray additive tank deletion. Other changes resulting from the VANTAGE 5 fuel reload do not affect the LOCA containment integrity analysis.

The effect that design changes to the reactor fuel assemblies can have on Containment Mass and Energy releases used to determine Containment Peak Pressure are dependent upon:

1) The change in core fluid volume as a result of the new fuel design.
2) Increase or decrease in core stored energy.
3) Effect of the new fuel design on reflood flooding rates as a result of core flow area or hydraulic resistance changes.

The VANTAGE 5 fuel design and the ANF 17x17 fuel design utilize a fuel rod smaller in diameter than the 15x15 OFA fuel which is modeled in the containment analysis documented in WCAP-11908. Therefore, the core stored energy is less than what is modeled in the WCAP-11908 analysis. The core volume is the same with 15x15 OFA fuel as with VANTAGE 5 and/or ANF fuel. The hydraulic resistance of the VANTAGE 5 fuel with the Intermediate Flow Mixing grids is larger than the hydraulic resistance of the 15x15 OFA fuel modeled in the analysis. The hydraulic resistance of the ANF 17x17 fuel is also larger than the hydraulic resistance of the 15x15 OFA fuel modeled in the analysis. The analysis, therefore, calculates conservatively high mass and energy releases to the containment. Thus, the containment analysis documented in WCAP-11908 bounds operation of Cook Nuclear Plant Unit 2 with a mixed ANF/VANTAGE 5 or full VANTAGE 5 core and the conclusions of WCAP-11908 remain valid.

5.2.7 Steam Generator Tube Rupture Analysis

The analysis for a Steam Generator Tube Rupture accident (SGTR) presented in the Cook Nuclear Plant Unit 2 UFSAR was performed to ensure that the offsite radiation doses remain below the limits based on the 10CFR100 guidelines.

A subsequent evaluation was performed and is documented in WCAP-11902 (Reference 25), Section 3.5, to determine the effect of increased power and revised temperature and pressure operation. This evaluation considered NSSS power levels up to 3600 MWt, a range of full power RCS vessel average temperatures between 547.0 °F and 581.3 °F, and RCS pressurizer pressures of 2250 psia or 2100 psia.

The evaluation also considered 10% average (15% peak) steam generator tube plugging, 15% auxiliary feedwater flow degradation, and 25 gpm charging flow imbalance. The other system performance degradation and fuel related changes considered in this report do not affect the SGTR accident analysis.

The primary thermal hydraulic parameters affecting the conclusion of the SGTR accident analysis are the extent of fuel failure, the primary to secondary break flow rate through the ruptured tube, and the mass released to the atmosphere from the steam generator with the ruptured tube. The UFSAR SGTR accident analysis and the WCAP-11902 evaluation are based on an assumption of 1% defective fuel, and an initial primary coolant activity corresponding to this amount of defective fuel. These assumptions will not be affected by the change to VANTAGE 5 fuel. The primary to secondary break flow rate and the mass release to the atmosphere are dependent upon the initial reactor and steam generator conditions of power. Since the range of operating conditions at Cook Nuclear Plant Unit 2 has been considered in WCAP-11902 and will not change due to the implementation of VANTAGE 5 fuel, it is concluded that the primary to secondary break flow rate and atmospheric steam release will not change due to the implementation of VANTAGE 5 fuel. Therefore, the consequences of a SGTR accident will not be increased by the implementation of VANTAGE 5 fuel and the SGTR accident evaluation in WCAP-11902 remains bounding.

5.3 LOCA Hydraulic Forces Analysis

5.3.1 Introduction

The purpose of the LOCA hydraulic forces analysis was to provide LOCA hydraulic forcing functions which were used in conjunction with the seismic analysis to verify the structural integrity of the core components for the proposed 17x17 VANTAGE 5 fuel reload, including the rerating program and peak steam generator tube plugging to 15% for Cook Nuclear Plant Unit 2 at the limiting primary fluid temperatures and pressures. The LOCA hydraulic forcing functions were

generated for the accumulator injection line break in the cold leg. The LOCA hydraulic forces analysis takes advantage of the elimination of large primary pipe ruptures (Reference 27) to reduce some of the expected increase in the magnitude of the peak forces which may occur due to the rerating program.

5.3.2 Method of Analysis

The method of analysis, to determine the LOCA hydraulic forcing functions, considers the accumulator injection line break at the reduced RCS primary temperatures, a core power of 3588 MWt, a peak steam generator tube plugging level of 15%, and a nominal RCS pressurizer pressure of 2250 psia. The computer codes that are used to evaluate the postulated LOCA are MULTIFLEX 1.0, LATFORC, and FORCE2. MULTIFLEX (Reference 28) is used to calculate the thermal hydraulics of the rector coolant system due to a postulated LOCA. LATFORC uses the pressure distribution in the downcomer annulus region calculated by MULTIFLEX to determine the lateral hydraulic forcing functions on the reactor vessel, core barrel and the thermal shield. FORCE2 uses the pressure transient in the reactor vessel calculated by MULTIFLEX to calculate the vertical forces on the vessel internals and core components.

5.3.3 Results

Results of the LOCA hydraulic forces analyses have shown that eliminating large pipe ruptures and analyzing reactor coolant branch line breaks partially offset the expected increases in the LOCA hydraulic forcing functions due to the reduced reactor coolant temperatures as proposed for the rerating program. Evaluations have shown that the LOCA hydraulic forcing functions from a double-ended guillotine break or a limited displacement break in the reactor coolant piping used in the structural integrity analyses (Reference 25) at current thermal conditions are still more limiting than the branch line LOCA hydraulic forcing functions at the reduced temperature conditions. Specifically, Reference 25 concluded that the peak horizontal forces from a 100 square inch reactor vessel inlet nozzle break remain limiting when compared to an accumulator injection line break. On this basis it was also concluded that the LOCA hydraulic forcing functions which were used as the bases for the original qualification of the reactor vessel, internals and loops remain bounding.

However, to specifically evaluate the structural integrity of the 17x17 VANTAGE 5 fuel, LOCA hydraulic forcing functions have been generated for the accumulator injection line break for the

rerating program to be used as input to determine the structural integrity of the core components. The evaluation of structural integrity for the core components can be found in Section 2.7 of this report which addresses seismic and LOCA considerations. This section provdies the evaluation and conclusions on the structural integrity of the 17x17 VANTAGE 5 fuel as a result of the accumulator injection line break LOCA hydraulic forcing functions calculated for the rerating program at reduced temperature conditions.

5.4 Steamline Break Mass and Energy Releases

This section addresses the impact of the complete transition of Cook Nuclear Plant Unit 2 from ANF 17x17 fuel to Westinghouse 17x17 VANTAGE 5 fuel on the Steamline Break Mass and Energy releases for both inside and outside containment.

5.4.1 Steamline Break Mass and Energy Releases Inside Containment

The Steamline Break Mass and Energy releases inside containment have been calculated to bound both Cook Nuclear Plant Unit 1 with 15x15 fuel and Cook Nuclear Plant Unit 2 VANTAGE 5 fuel. WCAP-11902, Supplement 1, Section S-3.3.4.1 documented this analysis which supports the Cook Nuclear Plant Unit 2 transition to VANTAGE 5 fuel, and includes the modified safety analysis assumptions as discussed in Section 5.1.3. The RCCA insertion time to dashpot assumed in the analysis was 2.4 seconds, which does not bound the 2.7 second time conservatively assumed for the VANTAGE 5 fuel. Also, the analysis did not consider a 10% Safety Injection Flow degradation. An evaluation has been performed which concludes that these differences would have an insiginificant effect on the calculated Mass and Energy releases. Thus, the analysis supports the transition to VANTAGE 5 fuel.

5.4.2 Steamline Break Mass and Energy Releases Outside Containment

The current Mass and Energy releases applicable for use in outside containment equipment qualification evaluation for Cook Nuclear Plant Unit 2 are documented in Reference 29 (Category 1). These releases included the effect of superheated steam for use in evaluation of the outside containment equipment qualification issues.

The Mass and Energy releases of Reference 30 have been evaluated for their applicability to the Cook Nuclear Plant Unit 2 VANTAGE 5 transition. This evaluation concludes that the Mass and Energy releases documented in Reference 30 will remain bounding for the transition of Cook

Nuclear Plant Unit 2 to VANTAGE 5 fuel, provided the following Technical Specifications and modified safety analysis assumptions/limitations are maintained. The outside containment Mass and Energy releases are insensitive to a 25 gpm charging flow imbalance.

- Maximum allowable NSSS power no greater than 3425 MWt.

- End-of-Cycle (EOC) Life most positive Moderator Density Coefficient (MDC) not more positive than 0.43 Δk/gm/cc. The Moderator Temperature Coefficient (MTC) as a function of vessel average temperature is shown in Figure 5.2

- Minimum shutdown margin of 1.6% Δk/k.

- Maximum allowable steamline isolation valve closure time no greater than 5.0 seconds (see NOTE below).

- The compensated nominal setpoint for low steamline pressure no less than 520 psig. This setpoint corresponds to the analysis setpoint of 379 psig.

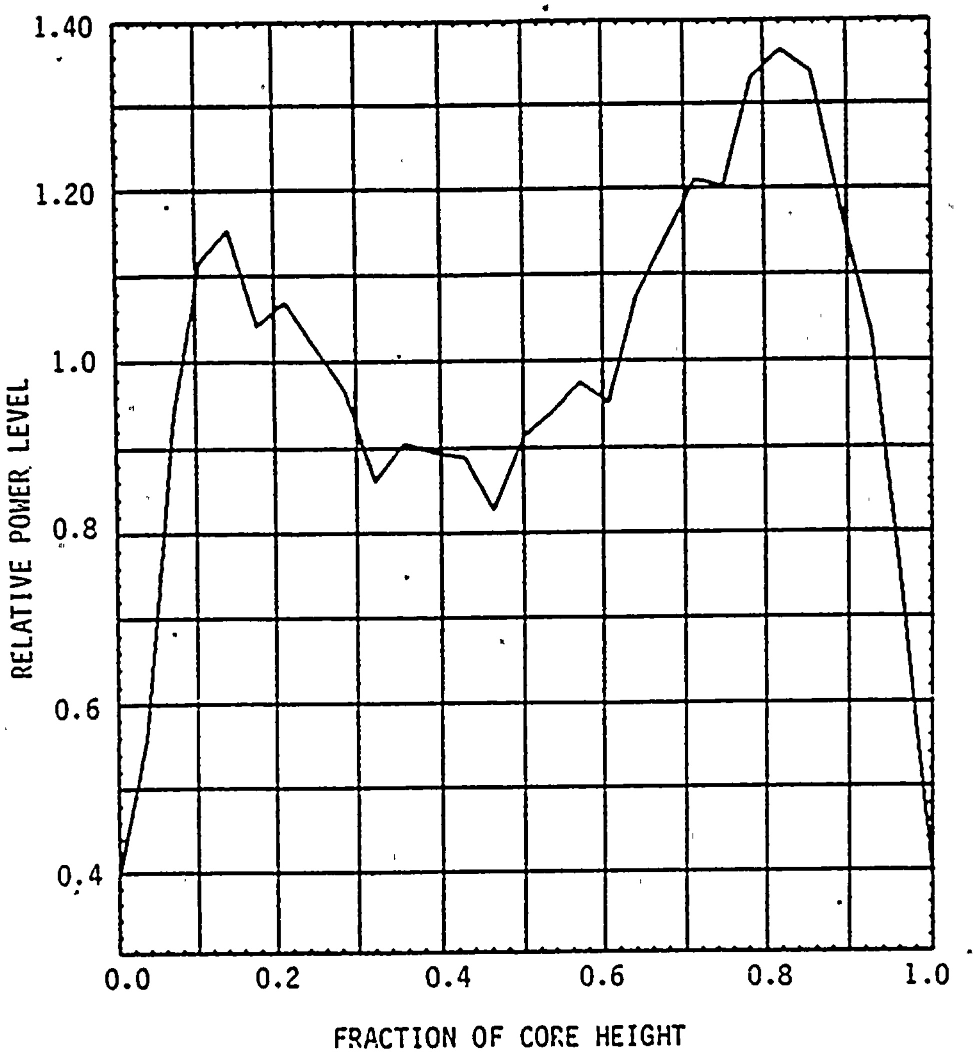
NOTE: A safety evaluation independent of the Cook Nuclear Plant Unit 2 VANTAGE 5 program has been performed to support an increase of 3.0 seconds in the steamline isolation valve closure time and related steamline isolation Engineered Safety Feature (ESF) response time (Reference 30).

The new superheated Mass and Energy releases to bound both Cook Nuclear Plant Unit 1 with 15x15 fuel and Cook Nuclear Plant Unit 2 with VANTAGE 5 fuel, including the modified safety analysis assumptions as discussed in Section 5.1.3 have been calculated by Westinghouse and were provided to AEPSC. The evaluation for determining the acceptability of these new superheated Mass and Energy releases for outside containment equipment qualification has not been completed for this report. The above Technical Specifications and modified safety analysis assumptions could be removed and the modified safety analysis assumptions as discussed in Section 5.1.3 could be supported at a later time, provided the new Mass and Energy releases are determined by AEPSC to be acceptable for outside containment equipment qualification.

5.4.3 Conclusions

The Cook Nuclear Plant Unit 2 transition to VANTAGE 5 fuel, including the modified safety analysis assumptions (Section 5.1.3) can be supported for the Mass and Energy releases inside containment.

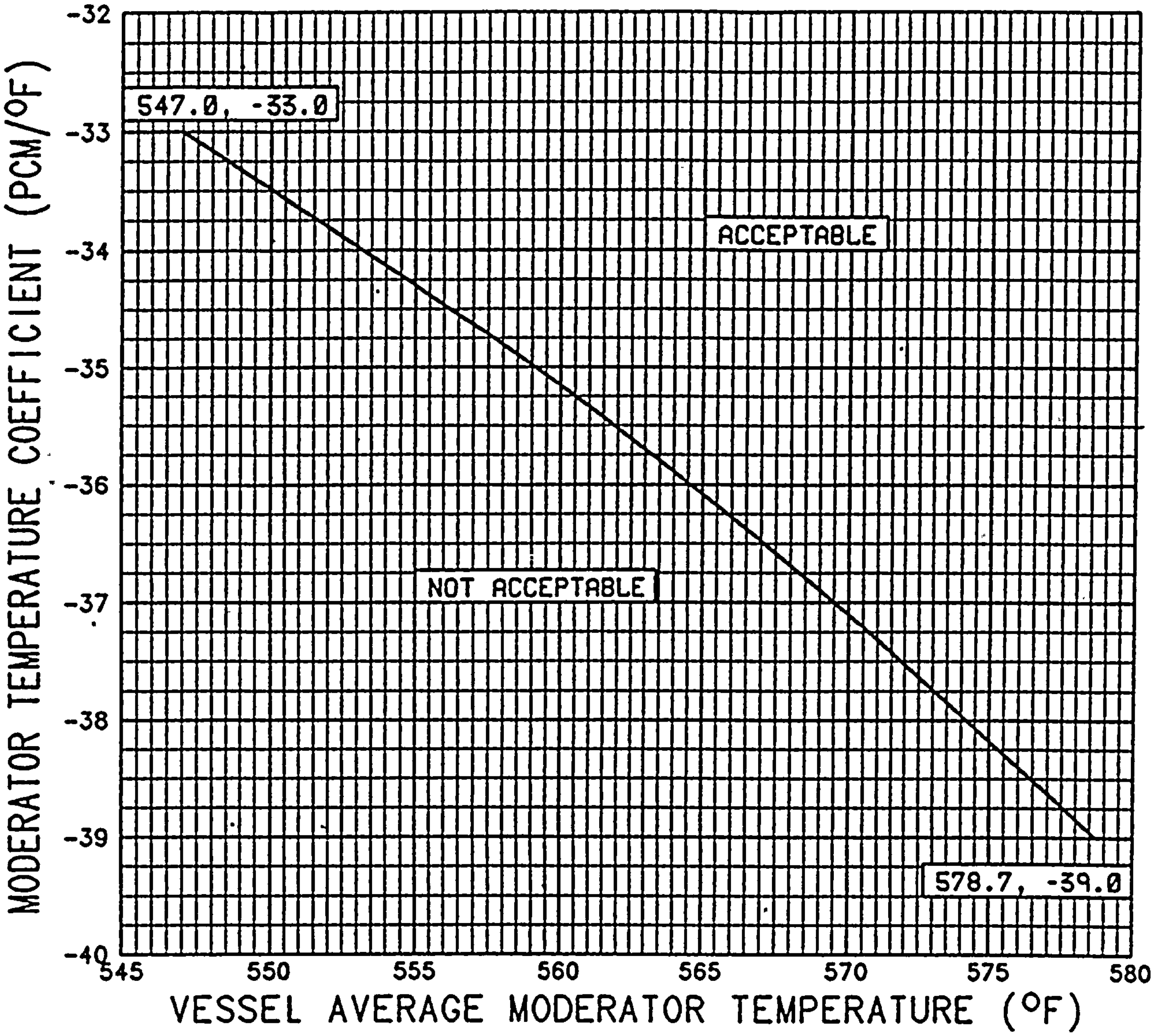
The current Mass and Energy releases outside containment as documented in Reference 5.29 will remain bounding for this report, provided the Technical Specifications limitations and modified safety analysis assumptions as noted in Section 5.4.2 are maintained.



COOK NUCLEAR PLANT UNIT 2

FIGURE 5.1

Design Axial Power Distribution
for non-OTΔT Transients
(WCAP-9500)



COOK NUCLEAR PLANT UNIT 2

FIGURE 5.2

Most Negative Moderator Temperature Coefficient Limit

6.0 SUMMARY OF TECHNICAL SPECIFICATIONS CHANGES

Table 6.1 presents a list of the Technical Specifications changes. The changes noted in Table 6.1 are given in the proposed Technical Specifications page changes in Appendix A.

TABLE 6.1

SUMMARY OF TECHNICAL SPECIFICATIONS CHANGES

SECTION, PAGE	CHANGE	REASON FOR CHANGE
1.0, pg I	Add COLR to index	COLR implementation
1.12a, pg 1-3	Add COLR	COLR implementation
Figure 2.1-1, pg 2-2	Revised safety limits	Reanalysis supports VANTAGE 5 reload
2.2.1, pg 2-5	Design flow change & trip setpoint	Change in design flow due to VANTAGE 5 fuel reload, RTDP implementation
Table 2.2-1, pg 2-7 & 2-8	Revise Overtemperature ΔT limits	Reanalysis supports VANTAGE 5 reload
Table 2.2-1, pg 2-9	Revise Overpower ΔT limits	Reanalysis supports VANTAGE 5 reload
2.1.1 Bases, pg B 2-1 & B 2-2	Update to bases	VANTAGE 5 fuel reload and COLR implementation (relocation of $F^N_{\Delta H}$)
2.1.1 Bases, pg B 2-4	Update to bases	VANTAGE 5 fuel reload and delete Cycle 6 specific information
2.1.1 Bases, pg B 2-5	Revise bases	Reanalysis supports VANTAGE 5 reload
2.1.1 Bases, pg B 2-7	Revise bases circuit breaker time	Reanalysis supports VANTAGE 5 reload
3/4.1.1.1, pg 3/4 1-1 & 1-2	Decrease shutdown margin	Reanalysis with reduced SDM
3/4.1.1.2, pg 3/4 1-3 & 1-3b	Decrease shutdown margin	Reanalysis with reduced SDM. Change to Westinghouse dilution accident methodology

TABLE 6.1
SUMMARY OF TECHNICAL SPECIFICATIONS CHANGES
(continued)

SECTION, PAGE	CHANGE	REASON FOR CHANGE
3/4.1.1.4, pg 3/4 1-5 & 3/4 1-6	MTC relocated to COLR & revised EOL limit	VANTAGE 5 fuel reload and COLR implementation (relocation of MTC)
3/4.1.1.5, pg 3/4 1-7	Minimum temperature for surveillance req.	Reanalysis with reduced temp
3/4.1.2.3, pg 3/4 1-11	Change ch. pump discharge head	Make consistent with the analysis
3/4.1.2.4, pg 3/4 1-12	Change ch. pump discharge head	Make consistent with the analysis
3/4.1.2.7, pg 3/4 1-15	Change 80 °F to 70 °F	Make spec consistent with the analysis limit
3/4.1.2.8, pg 3/4 1-16	Change volume from 5650 to 7715 gallons & change 80 °F to 70 °F	Make spec consistent with the VANTAGE 5 reload analysis limit to accommodate reduced rod worth and management flexibility
3/4.1.3.1, pg 3/4 1-19	Delete reference to Fig. 3.1-1	COLR implementation
3/4.1.3.4, pg 3/4 1-23	Change rod drop time from 2.2 to 2.7 sec Relocate steps withdrawn to COLR	Make spec consistent with the analysis limit & COLR implementation
3/4.1.3.5, pg 3/4 1-24	Relocate shutdown rod insertion limits to COLR	COLR implementation (relocation of shutdown rod insertion limits)

TABLE 6.1
SUMMARY OF TECHNICAL SPECIFICATIONS CHANGES
(continued)

SECTION, PAGE	CHANGE	REASON FOR CHANGE
3/4.1.3.6, pg 3/4 1-25	Relocate control rod insertion limits to COLR	COLR implementation (relocation of control rod insertion limits)
3/4.1.3.6, pg 3/4 1-26	Delete figure 3.1-1	COLR implementation
3/4.3.2.1, pg 3/4 2-1 & 2-3	Relocate axial flux difference limits to COLR	COLR implementation (relocation of AFD limits)
3/4.3.2.1, pg 3/4 2-4	Relocate axial flux difference allowable deviation Fig. to COLR	COLR implementation (relocation of AFD allowable deviation)
3/4.3.2.2, pg 3/4 2-5	Relocate F_Q limits to COLR	COLR implementation (relocation of F_Q limit)
3/4.3.2.2, pg 3/4 2-8, 2-8a & 2-8b	Relocate K(Z) & V(Z) figures to COLR	COLR implementation (relocation of F_Q limit)
3/4.3.2.3, pg 3/4 2-9	Relocate $F^N_{\Delta H}$ limits to COLR	COLR implementation (relocation of $F^N_{\Delta H}$ limit)
3/4.2.5.1, pg 3/4 2-15	Reformat DNB spec Change DNB parameter values and add low Tavg window	Adopt planned Cook Nuclear Plant Unit 1 spec format consistent with VANTAGE 5 reload
3/4.2.5.1, pg 3/4 2-16 & 2-17 & 2-18	Delete tables 3.2-1 and 3.2-2 Delete 3.2.5.2	Adopt planned Cook Nuclear Plant Unit 1 spec format Not required

TABLE 6.1
SUMMARY OF TECHNICAL SPECIFICATIONS CHANGES
(continued)

SECTION, PAGE	CHANGE	REASON FOR CHANGE
3/4.3.2.6, pg 3/4 2-19	Relocate F_Q limits to COLR Changed definition of F_Q	COLR implementation (relocation of F_Q limit) Westnghouse CAOC methodology
Table 3.3-2, pg 3/4 3-9 & 3-10	Changed and added RPS response times	Make consistent with the analysis limits
Table 3.4-4, pg 3/4 3-25	Change ESFAS setpoint	Make consistent with analysis
Table 3.3-5, pg 3/4 3-26 & 3/4 3-27 & 3/4 3-28	Changed ESF response time times	Make consistent with the analysis limits
3/4.4.1.2, pg 3/4 4-2 & 4-3	Reduce number of RCPs required operable in mode 3	Make consistent with the analysis limits
3/4.4.4, pg 3.4 4-6	Change water volume from 62% to 92%	Make consistent with the analysis limit
3/4.4.6.2, pg 3/4 4-15 & 3/4 4-16	Controlled leakage in terms of resistance	Consistent with analysis
3/4.5.1b, pg 3/4 5-1	Revise minimum contained borated water volume & min/max cover-pressure	Make consistent with analysis limits
3/4.5.2.f, pg 3/4 5-5	Revised SI pump performance	Reanalysis with degraded SI performance

TABLE 6.1
SUMMARY OF TECHNICAL SPECIFICATIONS CHANGES
(continued)

SECTION, PAGE	CHANGE	REASON FOR CHANGE
3/4.5.2.h, pg 3.4 5-6	Revised SI pump flow balance limits	Adopt limits similar to Cook Nuclear Plant Unit 1
3/4.5.5, pg 3/4 5-11	Reduce RWST min temp to 70 °F	Make spec consistent with analysis limit
3/4.1.1.1, pg B 3/4 1-1	Decrease shutdown margin	Reanalysis with reduced shutdown margin
B 3/4.1, pg B 3/4 1-3	Revise concentrations and volumes	Make spec consistent with analysis limits
B 3/4.2.1, pg B 3/4 2-1 & 2-2 & 2-3	Revise to reflect COLR implementation Changed to WCAP-8385	COLR implementation (relocation of AFD limits) Westinghouse methodology
B 3/4.2.2 & 3, pg B 3/4 2-4 thru 2-4b	Revised to reflect COLR implementation & VANTAGE 5 reload	VANTAGE 5 reload T-H analysis and COLR implementation (relocation of F_Q and $F^N_{\Delta H}$ limits)
B 3/4.2.5, pg B 3/4 2-5	Revise to reflect reduced temp DNB limit	Reanalysis with reduced temp
B 3/4.2.6, pg B 3/4 2-5	Revise to reflect CAOC control	Make spec consistent with analysis
B 3/4.5.5, pg B 3/4 5-3	Reduce RWST temp to 70 °F	Make spec consistent with the analysis limit
B 3/4.7.1, pg B 3/4 7-1	Reformat valve lift criteria	Make consistent with the analysis limit

TABLE 6.1
SUMMARY OF TECHNICAL SPECIFICATIONS CHANGES
(continued)

SECTION, PAGE	CHANGE	REASON FOR CHANGE
3.4.9.1, pg B 3/4 9-1	Delete reference to refueling reactivity calcs at 2000 ppm	Reanalysis of refueling reactivity at 2400 ppm boron
6.9.1.11, pg 6-18	Add COLR to section 6	COLR implementation

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APPENDIX A

TECHNICAL SPECIFICATIONS CHANGE PAGES

FOR THE

DONALD C. COOK NUCLEAR PLANT UNIT 2

TRANSITION TO 17x17 VANTAGE 5 FUEL

DEFINITIONS

CHANNEL FUNCTIONAL TEST

1.11 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the primary sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

b. Bistable channels - the injection of a simulated signal into the channel sensor to verify OPERABILITY including alarm and/or trip functions.

CORE ALTERATION

1.12 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

[INSERT] SEE ATTACHED INSERT B

SHUTDOWN MARGIN

1.13 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

IDENTIFIED LEAKAGE

1.14 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or

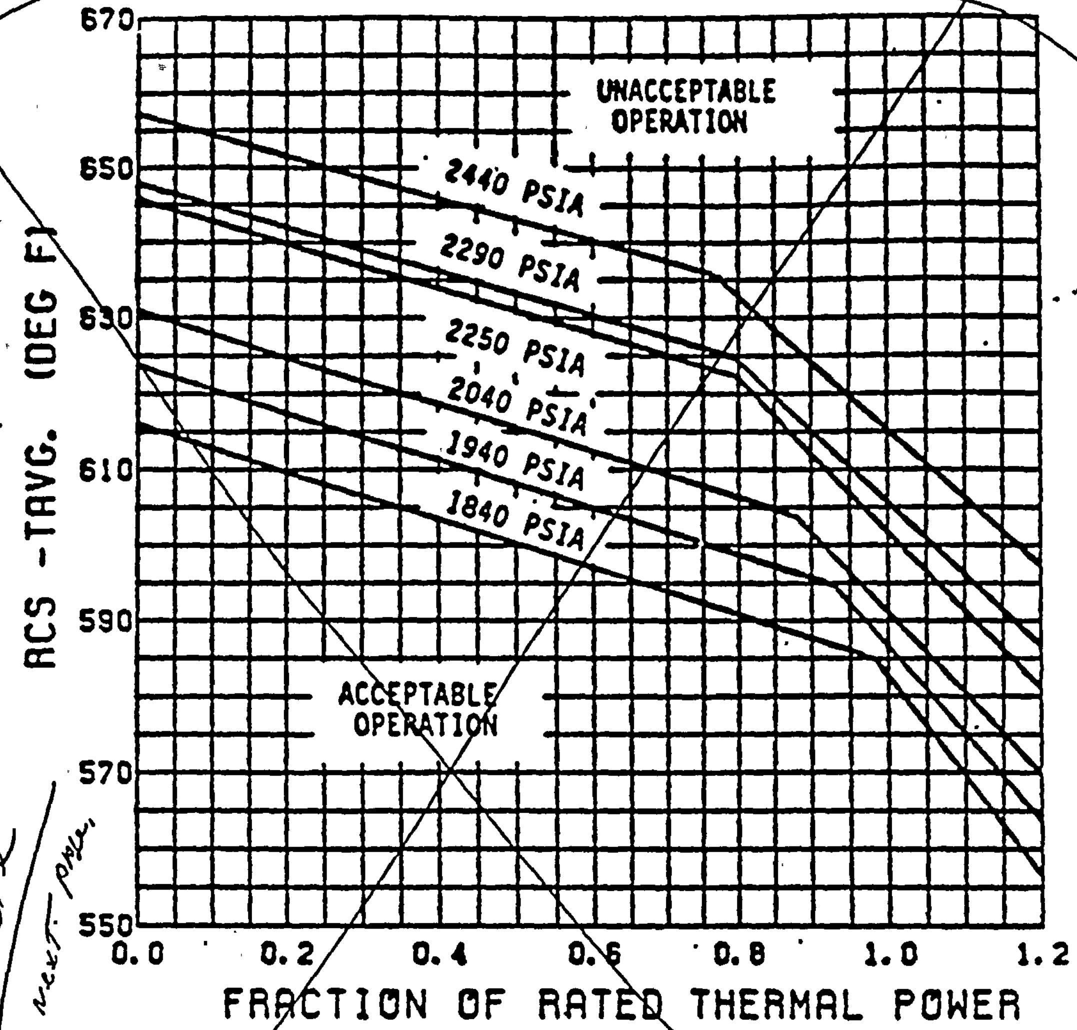
c. Reactor coolant system leakage through a steam generator to the secondary system.

INSERT B

<u>This is to be added to Section 1.0 - DEFINITIONS</u>

CORE OPERATING LIMITS REPORT	The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification ~~[illegible]~~ Unit operation within these operating limits is addressed in individual specifications.

6.9.1.11



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PRESSURE (PSIA)	BREAKPOINTS (FRACTION RATED THERMAL POWER, T AVG, DEG F)
1840	(0.00,616.2) , (0.98,585.1) , (1.20,556.5)
1940	(0.00,623.8) , (0.93,594.7) , (1.20,563.5)
2040	(0.00,631.0) , (0.88,603.8) , (1.20,569.6)
2250	(0.00,645.9) , (0.80,622.3) , (1.20,580.9)
2290	(0.00,647.9) , (0.80,624.5) , (1.20,586.5)
2440	(0.00,657.4) , (0.77,635.6) , (1.20,597.2)

FIGURE 2.1-1 Reactor Core Safety Limits - Four Loops in Operation

DESIGN FLOW = 91,600 gpm/Loop.

Description of Safety Limits

Pressure (psia)	Power (frac)	Tavg (°F)	Power (frac)	Tavg (°F)	Power (frac)	Tavg (°F)	Power (frac)	Tavg (°F)
1775	0.00	615.4	0.98	583.8	1.02	580.9	1.2	558.1
2000	0.00	631.8	0.86	605.8	0.96	597.5	1.2	568.5
2100	0.00	639.1	0.82	614.0	0.96	601.6	1.2	573.1
2250	0.00	649.2	0.72	628.6	0.98	605.2	1.2	580.4
2400	0.00	659.0	0.62	642.0	1.1	599.0	1.2	588.1

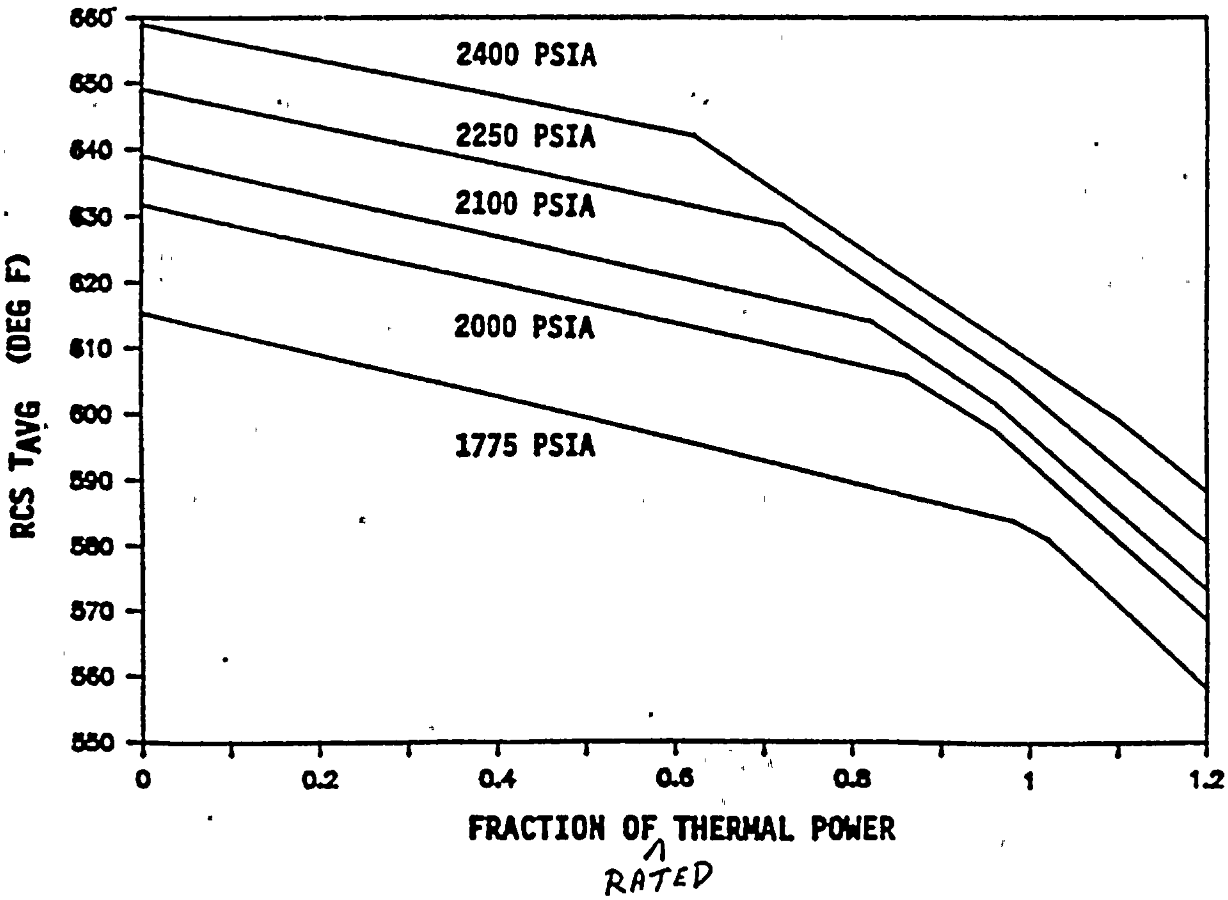

Figure 2.1-1 Reactor Core Safety Limits
Four Loops in Operation

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Range, Neutron Flux	Low Setpoint - ≤ 25% of RATED THERMAL POWER High Setpoint - ≤ 109% of RATED THERMAL POWER	Low Setpoint - ≤ 26% of RATED THERMAL POWER High Setpoint - ≤ 110% of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	≤ 5% of RATED THERMAL POWER with a time constant ≥ 2 seconds	≤ 5.5% of RATED THERMAL POWER with a time constant ≥ 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	≤ 5% of RATED THERMAL POWER with a time constant ≥ 2 seconds	≤ 5.5% of RATED THERMAL POWER with a time constant ≥ 2 seconds
5. Intermediate Range, Neutron Flux	≤ 25% of RATED THERMAL POWER	≤ 30% of RATED THERMAL POWER
6. Source Range, Neutron Flux	≤ 10^5 counts per second	≤ 1.3×10^5 counts per second
7. Overtemperature ΔT	See Note 1	See Note 3
8. Overpower ΔT	See Note 2	See Note 4
9. Pressurizer Pressure--Low	≥ 1950 psig	≥ 1940 psig
10. Pressurizer Pressure--High	≤ 2385 psig	≤ 2395 psig
11. Pressurizer Water Level--High	≤ 92% of instrument span	≤ 93% of instrument span
12. Loss of Flow	≥ 90% of design flow per loop*	89.0 ≥ 89.1% of design flow per loop*

*Design flow is ~~91,240~~ 91,600 gpm per loop.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

NOTE 1: Overtemperature $\Delta T \leq \Delta T_o \; [K_1 - K_2 \left[\frac{1+\tau_1 \cancel{8}\,s}{1+\tau_2 \cancel{8}\,s}\right] (T-T') + K_3(P-P') - f_1(\Delta I)]$

where: ΔT_o = Indicated ΔT at RATED THERMAL POWER

T = Average temperature, ^oF

T' = Indicated T_{avg} at RATED THERMAL POWER $\leq$ ~~574.1 °F~~ 576.0°F

P = Pressurizer Pressure, psig

P' = 2235 psig (indicated RCS nominal operating pressure)

$\frac{1+\tau_1 \cancel{8}\,s}{1+\tau_2 \cancel{8}\,s}$ = The function generated by the lead-lag controller for T_{avg} dynamic compensation

τ_1 & τ_2 = Time constants utilized in the lead-lag controller for T_{avg} τ_1 = ~~33~~ 28 secs, τ_2 = 4 secs.

$\cancel{8}\,s$ = Laplace transform operator

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

4 Loops in Operation

K_1 = ~~1.2590~~ 1.09

K_2 = ~~0.01374~~ 0.01331

K_3 = ~~0.000838~~ 0.00058

and f_1 (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for $q_t - q_b$ between ~~-31~~ -33 percent and ~~+3~~ +6 percent, f_1 (ΔI) = 0 (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER).

(ii) for each percent that the magnitude of ($q_t - q_b$) exceeds ~~-31~~ -33 percent, the ΔT trip setpoint shall be automatically reduced by ~~2.9~~ 3.5 percent of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of ($q_t - q_b$) exceeds ~~+3~~ +6 percent, the ΔT trip setpoint shall be automatically reduced by ~~2.2~~ 1.0 percent of its value at RATED THERMAL POWER.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Note 2: Overpower $\Delta T \leq \Delta T_o \, [K_4 - K_5 \left[\frac{\tau_3 s}{1+\tau_3 s}\right] T - K_6 \, (T - T'') - f_2(\Delta I)]$

where:

ΔT_o = Indicated ΔT at rated power

T = Average temperature, ^oF

T'' = Indicated T_{avg} at RATED THERMAL POWER $\leq$ ~~574.1~~ 576.0 ^oF.

K_4 = ~~1.078~~ 1.08

K_5 = $0.02/^oF$ for increasing average temperature and 0 for decreasing average temperature

K_6 = 0.00197 for $T > T''$; $K_6 = 0$ for $T \leq T''$

$\frac{\tau_3 s}{1+\tau_3 s}$ = The function generated by the rate lag controller for T_{avg} dynamic compensation

τ_3 = Time constant utilized in the rate lag controller for T_{avg} $\tau_3 = 10$ secs.

s = Laplace transform operator

$f_2(\Delta I)$ = ~~$f_1(\Delta I)$ as defined in Note 1 above~~ 0.0

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than ~~3.3~~ 1.3 percent ΔT span.

Note 4: The channel's maximum trip point shall not exceed its computed trip point by more than ~~2.6~~ 3.0 percent ΔT span.

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB. This relation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The DNB design basis is as follows: there must be at least a 95 percent probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (the XNB correlation in this application). The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the DNBR limit.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature below which the calculated DNBR is no less than the correlation DNBR limit value or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid. Uncertainties in primary system pressure, core temperature, core thermal power, primary coolant flow rate, and fuel fabrication tolerances have been included in the analyses from which Figure 2.1-1 is derived.

See insert C

SAFETY LIMITS

$F_{\Delta H}^{RTP}$ defined in the CORE OPERATING LIMITS REPORT.

BASES

The curves are based on a nuclear enthalpy rise hot channel factor, ~~$F^N_{\Delta H}$ of 1.49~~ $F_{\Delta H}^{RTP}$ and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = \cancel{1.49}\; F_{\Delta H}^{RTP}\, [1 + \cancel{0.2}\; PF_{\Delta H}\, (1-P)]$$ ~~(Westinghouse Fuel)~~

~~(Exxon Nuclear Company Fuel)~~

where P is the fraction of RATED THERMAL POWER.
and $PF_{\Delta H}$ is defined in the CORE OPERATING LIMITS REPORT

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f_1 (ΔI) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the setpoints to provide protection consistent with core safety limits.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Code for Nuclear Power Plant which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Reactor Coolant System piping, valves and fittings, are designed to ANSI B 31.1 1967 Edition, which permits a maximum transient pressure of 120% (2985 psig) of component design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3107 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

LIMITING SAFETY SYSTEM SETTINGS

BASES

The Power Range Negative Rate trip provides protection to ensure that the calculated DNBR is maintained above the design DNBR value for multiple control rod drop accidents. The analysis of a single control rod drop (OR SOME MULTIP ROD DROP accident indicates a return to full power may be initiated by the automatic control system in response to a continued full power turbine load demand or by the negative moderator temperature feedback. ~~A single control rod drop analysis with automatic rod control has not been performed for Cycle 6. The plant will be operated under the "Interim Criteria for Single Dropped Rod."~~

Intermediate and Source Range, Nuclear Flux

The Intermediate and Source Range, Nuclear Flux trips provide reactor core protection during reactor startup. These trips provide redundant protection to the low setpoint trip of the Power Range, Neutron Flux channels. The Source Range Channels will initiate a reactor trip at about 10^5 counts per second unless manually blocked when P-6 becomes active. The Intermediate Range Channels will initiate a reactor trip at a current level proportional to approximately 25 percent of RATED THERMAL POWER unless manually blocked when P-10 becomes active. No credit was taken for operation of the trips associated with either the Intermediate or Source Range Channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Protection System.

Overtemperature ΔT

The Overtemperature ΔT trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds), and pressure is within the range between the High and Low Pressure reactor trips. This setpoint includes corrections for changes in density and heat capacity of water with temperature and dynamic compensation for piping delays from the core to the loop temperature detectors. This reactor trip limit is always below the core safety limit as shown in Figure 2.1-1. If axial peaks are more severe than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced according to the notations in Table 2.2-1.

2.1 SAFETY LIMITS

BASES

Insert C

2.1.1 REACTOR CORE

The restrictions of this Safety Limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the WRB-2 correlation and W-3 correlation for conditions outside the range of WRB-2. The DNB correlations have been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio, (DNBR), is defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, and is indicative of the margin to DNB.

The DNB design basis is as follows: there must be at least a 95 percent probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (the WRB-2 correlation for Vantage-5 fuel, and the W-3 correlation for ANF fuel and conditions which fall outside the range of applicability of the WRB-2). The correlation DNBR limits are established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the DNBR limit (1.17 for WRB-2 and 1.3 for the W-3).

In meeting the DNB design basis, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are statistically combined with the DNBR correlation statistics such that there is at least a 95% probability with a 95% confidence level that the minimum DNBR for the limiting rod is greater than or equal to a calculated design limit DNBR. The uncertainties in the above plant parameters are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the DNBR correlation statistics establishes a design DNBR value which must be met in plant safety analyses using values of input parameters without uncertainties. For D. C. Cook Unit 2, the design DNBR values are 1.23 and 1.22 for Vantage-5 fuel typical and

INSERT C CONTINUED

thimble cells, respectively, and 1.39 and 1.36 for typical and thimble cells for the ANF fuel. In addition, margin has been maintained in both fuel types by performing safety analyses to a safety analysis limit DNBR. The margin between the design and safety analysis limit DNBR is used to offset known DNBR penalties (i.e. transition core penalties, rod bow, etc.) and provide DNBR margin for operating and design flexibility.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature below which the calculated DNBR is no less than the design DNBR limit value or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid.

Overpower ΔT OF OVERPOWER DELTA-T TRIP

The Overpower ΔT reactor trip provides assurance of fuel integrity, e.g., no melting, under all possible overpower conditions, limits the required range for Overtemperature ΔT protection, and provides a backup to the High Neutron Flux trip. The setpoint includes corrections for changes in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System. ~~If axial peaks are more severe than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced according to the notations in Table 2.2-1.~~

Pressurizer Pressure

The Pressurizer High and Low Pressure trips are provided to limit the pressure range in which reactor operation is permitted. The High Pressure trip is backed up by the pressurizer code safety valves for RCS overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The High Pressure trip provides protection for a Loss of External Load event. The Low Pressure trip provides protection by tripping the reactor in the event of a loss of reactor coolant pressure.

Pressurizer Water Level

The Pressurizer High Water Level trip ensures protection against Reactor Coolant System overpressurization by limiting the water level to a volume sufficient to retain a steam bubble and prevent water relief through the pressurizer safety valves. ~~No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.~~ The PRESSURIZER High WATER LEVEL TRIP PRECLUDES WATER RELIEF FOR THE UNCONTROLLED ROD WITHDRAWAL AT POWER EVENT.

Undervoltage and Underfrequency - Reactor Coolant Pump Busses

The Undervoltage and Underfrequency Reactor Coolant Pump bus trips provide reactor core protection against DNB as a result of loss of voltage or underfrequency to more than one reactor coolant pump. The specified set points assure a reactor trip signal is generated before the low flow trip set point is reached. Time delays are incorporated in the underfrequency and undervoltage trips to prevent spurious reactor trips from momentary electrical power transients. For undervoltage, the delay is set so that the time required for a signal to reach the reactor trip breakers following the simultaneous trip of two or more reactor coolant pump bus circuit breakers shall not exceed ~~0.9~~ 1.2 seconds. For underfrequency, the delay is set so that the time required for a signal to reach the reactor trip breakers after the underfrequency trip set point is reached shall not exceed 0.3 seconds.

Turbine Trip

A Turbine Trip causes a direct reactor trip when operating above P-7. Each of the turbine trips provide turbine protection and reduce the severity of the ensuing transient. No credit was taken in the accident analyses for operation of these trips. Their functional capability at the specified trip settings is required to enhance the overall reliability of the Reactor Protection System.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - ~~STANDBY, STARTUP, AND POWER OPERATION~~

T_{avg} GREATER THAN 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be ≥ ~~2.0%~~ 1.6 Δk/k.

APPLICABILITY: MODES 1, 2*, ~~and~~ ~~3~~ 3 AND 4

ACTION:

With the SHUTDOWN MARGIN < ~~2.0%~~ 1.6% Δk/k, immediately initiate and continue boration at ≥ 10 gpm of 20,000 ppm boric acid solution or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be ≥ ~~2.0%~~ 1.6% Δk/k:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable control rod(s).

b. When in MODES 1 or 2#, at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6.

c. When in MODE 2##, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6.

---

*See Special Test Exception 3.10.1

\# With $K_{eff} \geq 1.0$

\## With $K_{eff} < 1.0$

SURVEILLANCE REQUIREMENTS (Continued)

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the control banks at the maximum insertion limit of Specification 3.1.3.6.

e. When in MODE 3, at least once per 24 hours by consideration of the following factors:

1. Reactor coolant system boron concentration,
2. Control rod position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1% Δk/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.e, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

1.6%

4.1.1.1.3 Prior to blocking ESF Functional Units in accordance with footnotes # and ## of Table 3.3-3, SHUTDOWN MARGIN shall be determined to be greater than or equal to ~~1.0%~~ Δk/k by consideration of the factors of 4.1.1.1.1.e above. The Reactor Coolant System average temperature used in making this SHUTDOWN MARGIN determination shall be less than or equal to 350°F. This SHUTDOWN MARGIN shall be maintained at all times when the ESF functions are blocked in MODE 3.

REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be:

~~a. In MODE 4:~~

1. Greater than or equal to 2.0% Δk/k when operating with one or more Reactor Coolant Loops in accordance with Specification 3.4.1.3.

2. Greater than the value shown in Figure 3.1-3 when operating with no Reactor Coolant Loops but one or more Residual Heat Removal Loops in accordance with Specification 3.4.1.3.

~~b. In MODE 5:~~

1. Greater than or equal to 1.0% Δk/k when operating ~~with one or more Reactor Coolant Loops~~ in accordance with Specification 3.4.1.3.

2. Greater than the value shown in Figure 3.1-3 when operating with no Reactor Coolant Loops but one or more Residual Heat Removal Loops in accordance with Specification 3.4.1.3.

APPLICABILITY: MODES 4 and 5

ACTION:

With SHUTDOWN MARGIN less than the above limits, immediately initiate and continue boration at greater than or equal to 10 gpm of 20,000 ppm boric acid solution or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to the above limits:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable control rod(s).

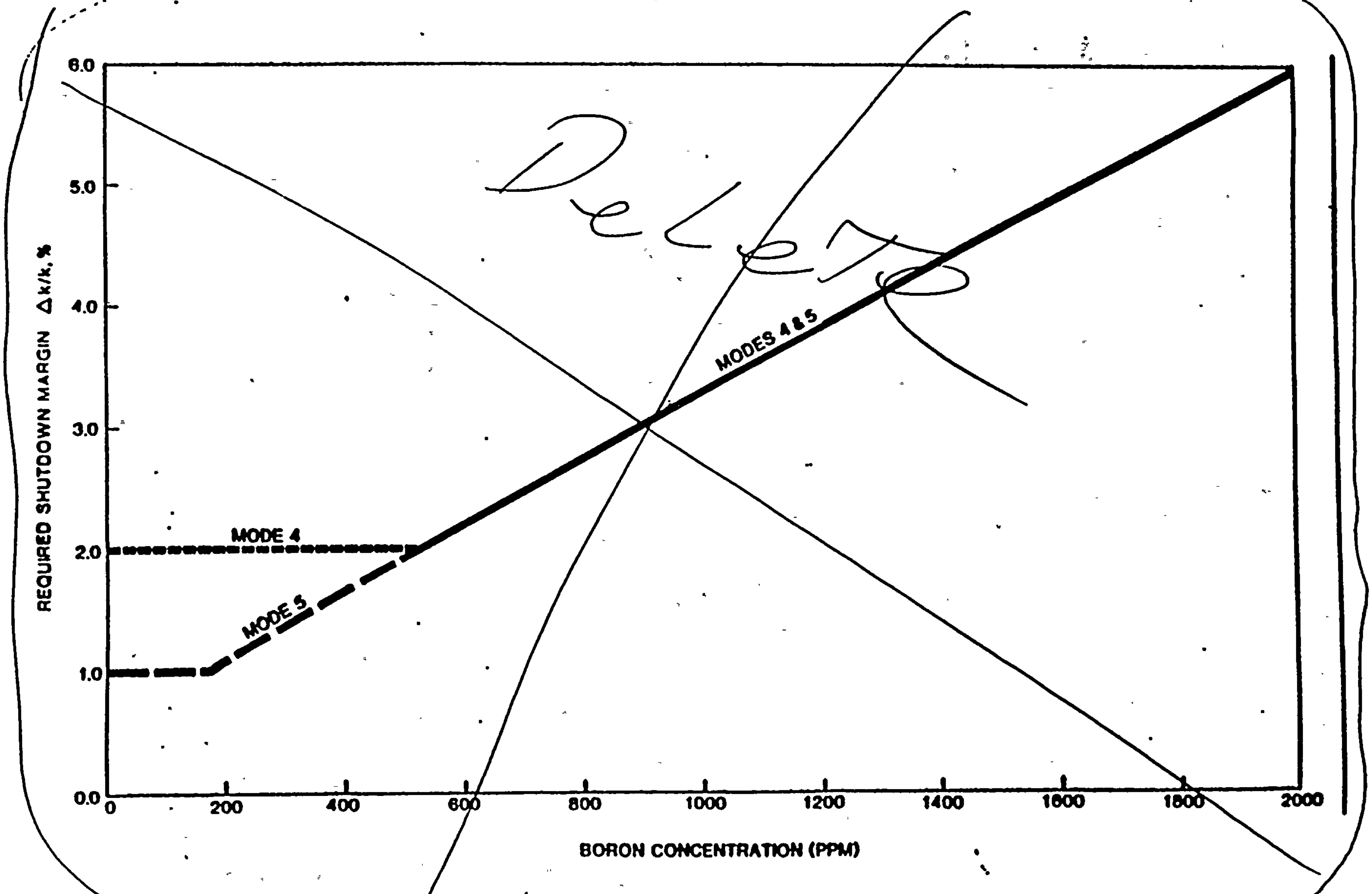

FIGURE 3.1-3 REQUIRED SHUTDOWN MARGIN

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.4 The moderator temperature coefficient (MTC) shall be ~~:~~ within the limits specified in the Core Operating Limits Report (COLR). The maximum
~~a. Within the region of acceptable operation in Figure 3.1-2, and~~
upper limit shall be less than or equal to the values in Figure 3.1-2
~~b. Less negative than -3.5×10^{-4} Δk/k/°F for the all rods withdrawn, end of cycle life (EOL), RATED THERMAL POWER condition.~~

APPLICABILITY: Beginning of Cycle Life (BOL) limit ~~Specification 3.1.1.4.a~~ - MODES 1 and 2* only#
End of Cycle Life (EOL) limit ~~Specification 3.1.1.4.b~~ - MODES 1, 2 and 3 only#

ACTION:

a. With the MTC more positive than the BOL limit specified in the COLR, ~~of 3.1.1.4.a above~~:

1. Establish and maintain control rod withdrawal limits sufficient to restore the MTC to within its limits within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6.

2. Maintain the control rods within the withdrawal limits established above until subsequent measurement verifies that the MTC has been restored to within its limit for the all rods withdrawn condition.

3. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 10 days describing the value of the measured MTC, the interim control rod withdrawal limits and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

b. With the MTC more negative than the EOL limit specified in the COLR ~~of 3.1.1.4.b above,~~ be in HOT SHUTDOWN within 12 hours.

---

* With K_{eff} greater than or equal to 1.0

\# See Special Test Exception 3.10.3

4.1.1.4.1 The MTC shall be determined to be within its limits by confirmatory measurements. Measured MTC values shall be extrapolated and/or compensated to permit direct comparison with the above limits.

4.1.1.4.2 The MTC shall be determined to be within its limits during each fuel cycle as follows:

a) The MTC shall be measured and compared to the BOL limit ~~of Specification 3.1.1.4.a, above~~, prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b) The MTC shall be measured at any THERMAL POWER within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm and the ~~extrapolated MTC~~ value compared to the ~~EOL limi~~t. In the event this comparison indicates that the MTC will be more negative than the EOL limit, the MTC shall be remeasured at least once per 14 EFPD during the remainder of the fuel cycle and the MTC value compared to the EOL limit.

measured

specified in the COLR

300 PPM surveillance limit specified in the COLR.

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.1.1.5 The Reactor Coolant System lowest operating loop temperature (T_{avg}) shall be $\geq$ 541°F.

APPLICABILITY: MODES 1 and 2#.

ACTION:

With a Reactor Coolant System operating loop temperature (T_{avg}) < 541°F, restore (T_{avg}) to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.5 The Reactor Coolant System temperature (T_{avg}) shall be determined to be $\geq$ 541°F:

a. Within 15 minutes prior to achieving reactor criticality, and

b. At least once per 30 minutes when the reactor is critical and the Reactor Coolant System T_{avg} is less than ~~551°F~~ 545°F with the T_{avg}-T_{ref} Deviation Alarm not reset.

#With $K_{eff} \geq 1.0$.

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 One charging pump in the boron injection flow path required by Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency bus.

APPLICABILITY: MODES 5 and 6.

ACTION:

a. With no charging pump OPERABLE, suspend all* operations involving CORE ALTERATIONS or positive reactivity changes.

b. With more than one charging pump OPERABLE or with a safety injection pump(s) OPERABLE when the temperature of any RCS cold leg is less than or equal to 152°F, unless the reactor vessel head is removed, remove the additional charging pump(s) and the safety injection pump(s) motor circuit breakers from the electrical power circuit within one hour.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.1.2.3.1 The above required charging pump shall be demonstrated OPERABLE by verifying that, on recirculation flow, the pump develops a discharge pressure of greater than or equal to ~~2390 psig~~ 2290 psid when tested pursuant to Specification 4.0.5.

4.1.2.3.2 All charging pumps and safety injection pumps, excluding the above required OPERABLE charging pump, shall be demonstrated inoperable by verifying that the motor circuit breakers have been removed from their electrical power supply circuits at least once per 12 hours, except when:

a. The reactor vessel head is removed, or

b. The temperature of all RCS cold legs is greater than 152°F.

---

* For purposes of this specification, addition of water from the RWST does not constitute a positive reactivity addition provided the boron concentration in the RWST is greater than the minimum required by Specification 3.1.2.7.b.2.

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% $\Delta k/k$ at 200°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4 At least two charging pumps shall be demonstrated OPERABLE by verifying, that on recirculation flow, each pump develops a discharge pressure of $\geq$ ~~2405 psig~~ 2290 psid when tested pursuant to Specification 4.0.5.

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.7 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A boric acid storage system and associated heat tracting with:

1. A minimum usable borated water volume of 4300 gallons,
2. Between 20,000 and 22,500 ppm of boron, and
3. A minimum solution temperature of 145°F.

b. The refueling water storage tank with:

1. A minimum usable borated water volume of 90,000 gallons,
2. A minimum boron concentration of 2400 ppm, and
3. A minimum solution temperature of ~~80°F.~~ 70°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes* until at least one borated water source is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.7 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration of the water,
2. Verifying the contained borated water volume, and
3. Verifying the boric acid storage tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water.

*For purposes of this specification, addition of water from the RWST does not constitute a dilution activity provided the boron concentration in the RWST is greater than or equal to the minimum required by Specification 3.1.2.7.b.2.

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.8 Each of the following borated water sources shall be OPERABLE:

a. A boric acid storage system and associated heat tracing with:

   1. A minimum usable borated water volume of ~~5650~~ 7715 gallons,

   2. Between 20,000 and 22,500 ppm of boron, and

   3. A minimum solution temperature of 145°F.

b. The refueling water storage tank with:

   1. A minimum contained volume of 350,000 gallons of water,

   2. Between 2400 and 2600 ppm of boron, and

   3. A minimum solution temperature of ~~80°F.~~ 70°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the boric acid storage system inoperable, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 1% Δk/k at 200°F; restore the boric acid storage system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water storage tank inoperable, restore the tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.8 Each borated water source shall be demonstrated OPERABLE:

LIMITING CONDITION FOR OPERATION (Continued)

c) A power distribution map is obtained from the movable incore detectors and $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours, and

d) Either the THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within one hour and within the next 4 hours the high neutron flux trip setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER, or

e) The remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod within one hour while maintaining the rod sequence and insertion limits ~~of Figure 3.1-1~~ as specified in the CORE OPERATING LIMITS REPORT; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full length rod shall be determined to be within the group demand limit by verifying the individual rod positions at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then verify the group positions at least once per 4 hours.

4.1.3.1.2 Each full length rod not fully inserted shall be determined to be OPERABLE by movement of at least 8 steps in any one direction at least once per 31 days.

REACTIVITY CONTROL SYSTEMS

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

As specified in The Core Operating Limits Report (COLR)

3.1.3.4 The individual full length (shutdown and control) rod drop time from the fully withdrawn position ~~(228 steps)~~ shall be less than or equal to ~~2.2~~ 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to 541°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

With the drop time of any full length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The rod drop time of full length rods shall be demonstrated through measurement prior to entering MODE 2:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and

c. At least once per 18 months.

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown rods shall be LIMITED IN PHYSICAL INSERTION AS SPECIFIED IN THE CORE OPERATING LIMITS REPORT (COLR) ~~fully withdrawn (228 steps)~~.

APPLICABILITY: MODES 1* and 2*#

ACTION:

With a maximum of one shutdown rod INSERTED BEYOND THE INSERTION LIMIT SPECIFIED IN THE COLR ~~not fully withdrawn~~, except for surveillance testing pursuant to Specification 4.1.3.1.2, within one hour either:

a. RESTORE THE ROD TO WITHIN THE INSERTION LIMIT SPECIFIED IN THE COLR, OR ~~Fully withdraw the rod, or~~

b. Declare the rod to be inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown rod shall be determined to be WITHIN THE INSERTION LIMIT SPECIFIED IN THE COLR: ~~fully withdrawn:~~

a. Within 15 minutes prior to withdrawal of any rods in control banks A, B, C or D during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

*See Special Test Exceptions 3.10.2 and 3.10.3
#With K_{eff} greater than or equal to 1.0

REACTIVITY CONTROL SYSTEMS

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as ~~shown in Figure 3.1-1.~~ specified in the Core Operating Limits Report (COLR)

APPLICABILITY: MODES 1* and 2*#.

ACTION:

With the control banks inserted beyond the ~~above~~ insertion limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, either:

a. Restore the control banks to within the limits within two hours, or

b. Reduce THERMAL POWER within two hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the group position using the ~~above figure, or~~ insertion limits specified in the COLR, or

c. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each control bank shall be determined to be within the insertion limits at least once per 12 hours except during time intervals when the Rod Insertion Limit Monitor is inoperable, then verify the individual rod positions at least once per 4 hours.

* See Special Test Exceptions 3.10.2 and 3.10.3.

\# With K_{eff} greater than or equal to 1.0.

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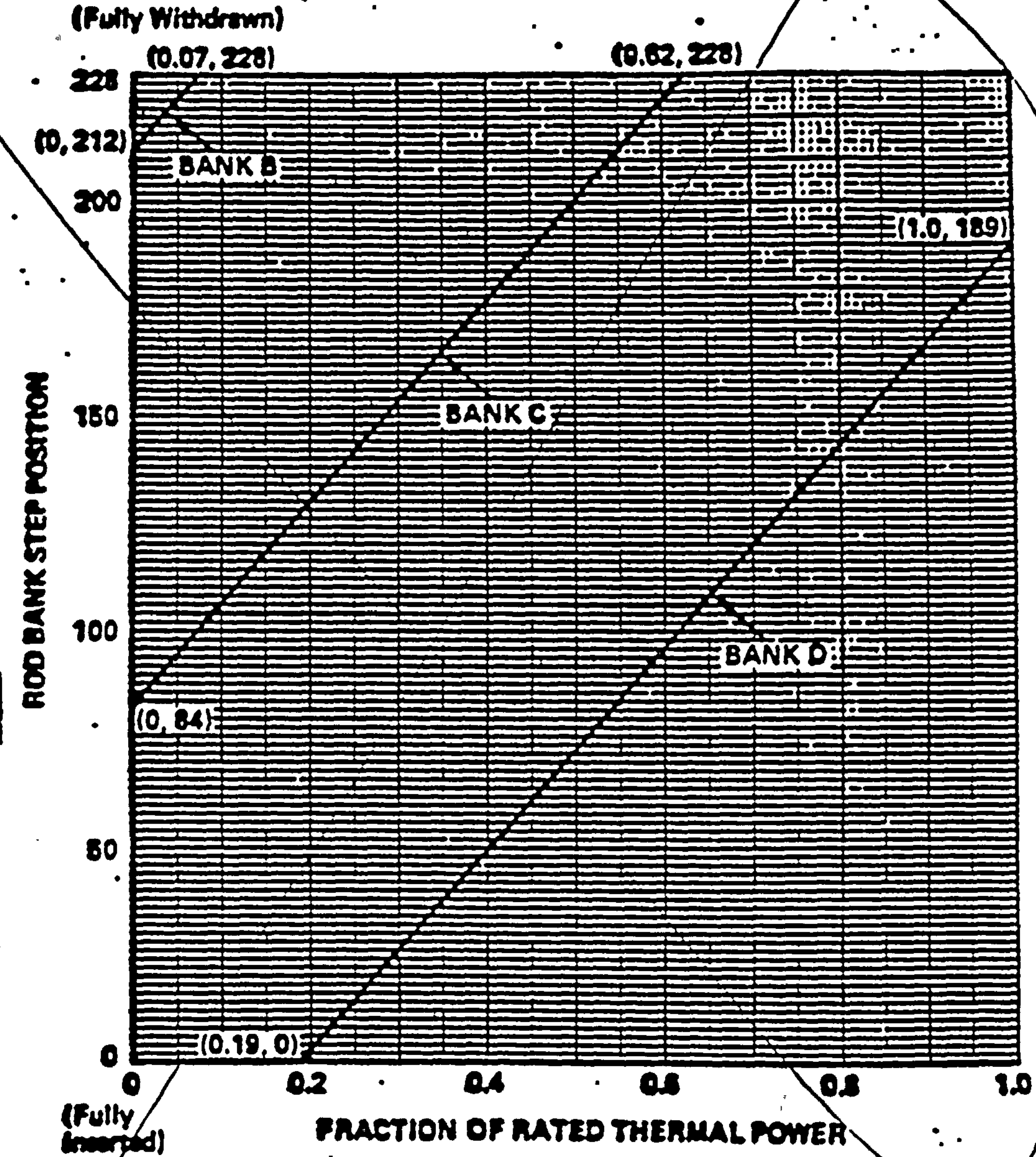

Figure 3.1-1
ROD BANK INSERTION LIMITS VERSUS THERMAL POWER
FOUR LOOP OPERATION

AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the target band ~~(±5% or ±3%)~~ ~~(flux difference units)~~ about a target flux difference. The TARGET BAND IS SPECIFIED IN THE CORE OPERATING LIMITS REPORT (COLR

APPLICABILITY: MODE 1 above 50% RATED THERMAL POWER*

ACTION:

a. With the indicated AXIAL FLUX DIFFERENCE outside of the target band about the target flux difference and with THERMAL POWER:

1. Above 90% or 0.9 x APL (whichever is less) of RATED THERMAL POWER, within 15 minutes:

   a) Either restore the indicated AFD to within the target band limits, or

   b) Reduce THERMAL POWER to less than 90% or 0.9 x APL (whichever is less) of RATED THERMAL POWER.

2. Between 50% and 90% or 0.9 x APL (whichever is less) of RATED THERMAL POWER:

   a) POWER OPERATION may continue provided:

      1) The indicated AFD has not been outside of the target band for more than 1 hour penalty deviation cumulative during the previous 24 hours, and

      2) The indicated AFD is within the limits ~~shown on Figure 3.2-1~~ Specified in the COLR. Otherwise, reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

   b) Surveillance testing of the Power Range Neutron Flux Channels may be performed pursuant to Specification 4.3.1.1.1 provided the indicated AFD is maintained within the limit ~~of Figure 3.2-1~~ Specified in the COLR. A total of 16 hours operation may be accumulated with the AFD outside of the target band during this testing without penalty deviation.

*See Special Test Exception 3.10.2

SURVEILLANCE REQUIREMENTS (continued)

4.2.1.2 The indicated AFD shall be considered outside of its target band when at least 2 of 4 or 2 of 3 OPERABLE excore channels are indicating the AFD to be outside the target band. Penalty deviation outside of the target band shall be accumulated on a time basis of:

a. A penalty deviation of one minute for each one minute of POWER OPERATION outside of the target band at THERMAL POWER levels equal to or above 50% of RATED THERMAL POWER, and

b. A penalty deviation of one half minute for each one minute of POWER OPERATION outside of the target band at THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER.

4.2.1.3 The target axial flux difference for the OPERABLE excore channels shall be determined in conjunction with the measurement of APL as defined in Specification 4.2.6.2. The provisions of Specification 4.0.4 are not applicable.

4.2.1.4 The axial flux difference target band about the target axial flux difference shall be determined in conjunction with the measurement of APL as defined in Specification 4.2.6.2. The allowable values of the target band are ~~±5% or ±3%. Redefinition of the target band from ±3% to ±5% between determinations of the target axial flux difference is allowed when appropriate redefinitions of APL are made. Redefinition of the target band from ±5% to ±3% is allowed only in conjunction with the determination of a new target axial flux difference.~~ The provisions of Specification 4.0.4 are not applicable.

Specified in the COLR

FIGURE 3.2-1 ALLOWABLE DEVIATION FROM TARGET FLUX DIFFERENCE

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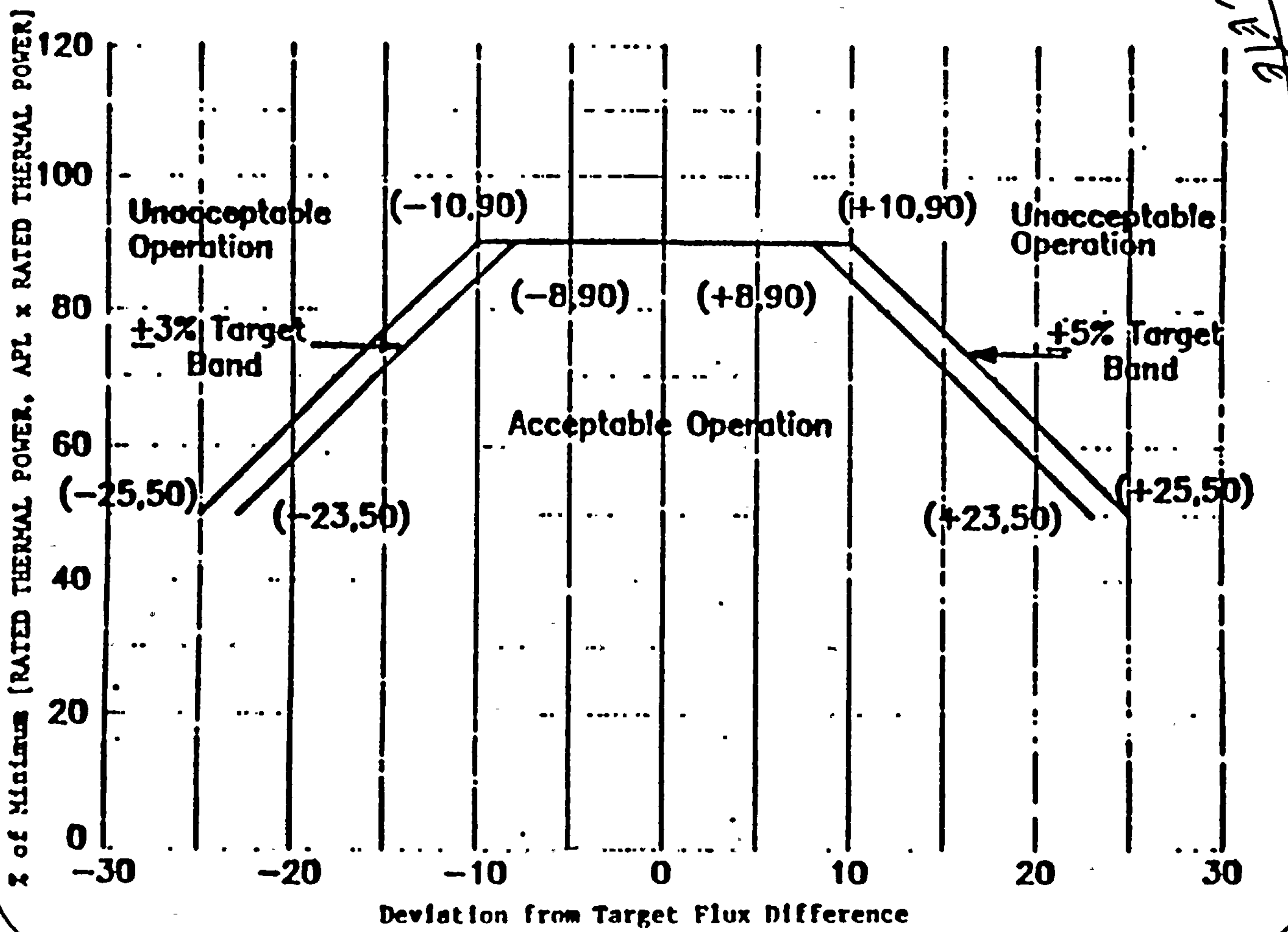

HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$ shall be limited by the following relationships:

~~Westinghouse Fuel~~ F_Q^{RTP} ~~Exxon Nuclear Co. Fuel~~

$F_Q(Z) \leq \frac{\cancel{[1.97]}}{P} [K(Z)]$ ~~$F_Q(Z) \leq \frac{[2.10]}{P} [K(Z)]$~~ $P > 0.5$

$F_Q(Z) \leq \cancel{[3.94]}\ \frac{F_Q^{RTP}}{0.5} [K(Z)]$ ~~$F_Q(Z) \leq [4.20] [K(Z)]$~~ $P \leq 0.5$

- $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

- $F_Q(Z)$ is the measured hot channel factor including a 3% manufacturing tolerance uncertainty and a 5% measurement uncertainty.

- K(Z) is the ~~function obtained from Figure 3.2-2 for Westinghouse fuel and Figure 3.2-2(a) for Exxon Nuclear Company fuel~~. NORMALIZED F_Q (Z) AS A FUNCTION OF CORE HEIGHT SPECIFIED IN THE COLR

APPLICABILITY: MODE 1

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit.

b. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a, above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

- F_Q^{RTP} The F_Q LIMIT AT RATED THERMAL POWER SPECIFIED IN THE CORE OPERATING LIMITS REPORT (COLR)

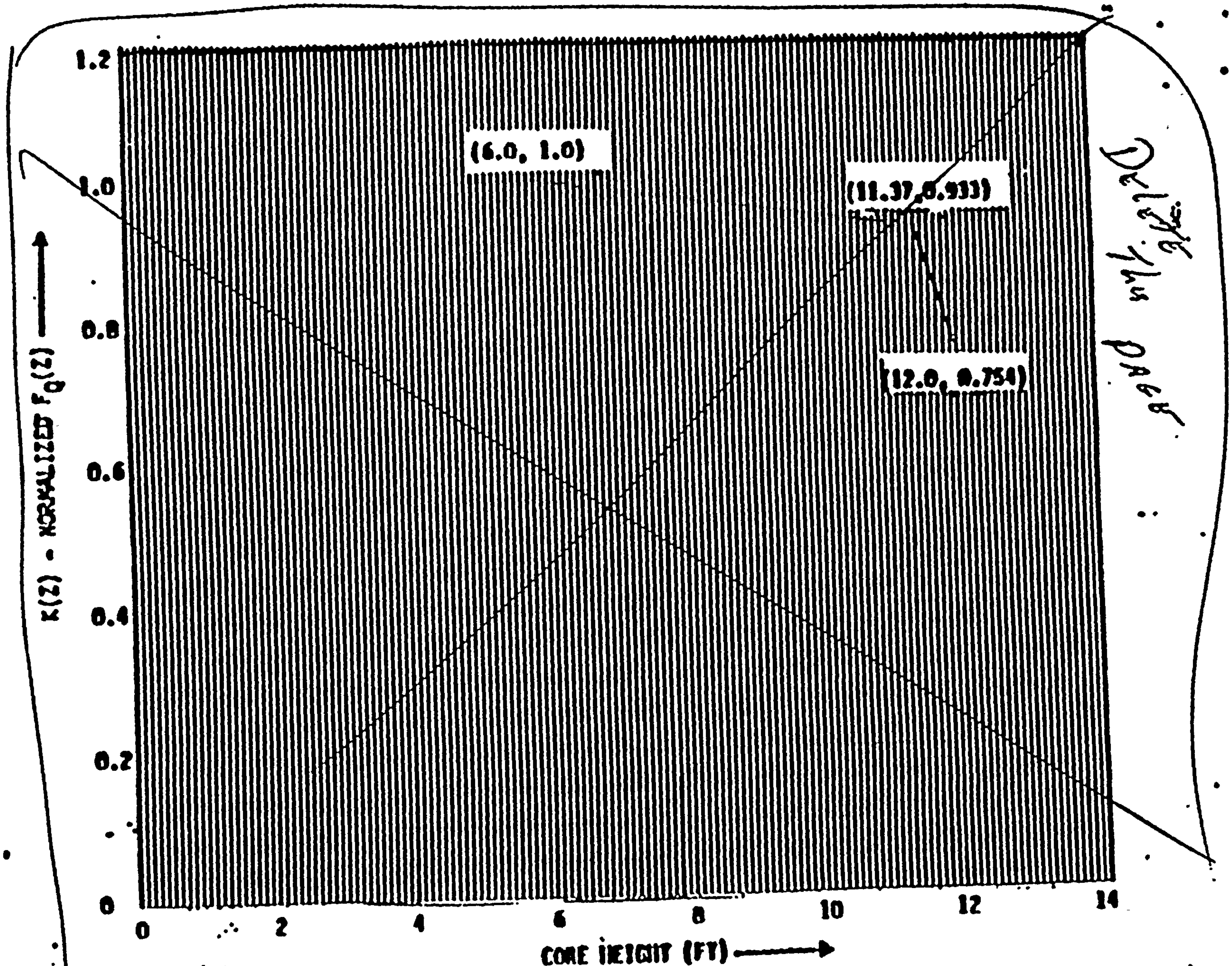

FIGURE 3.2-2 D. C. COOK UNIT 2 K(Z)-NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT

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D. C. COOK - UNIT 2

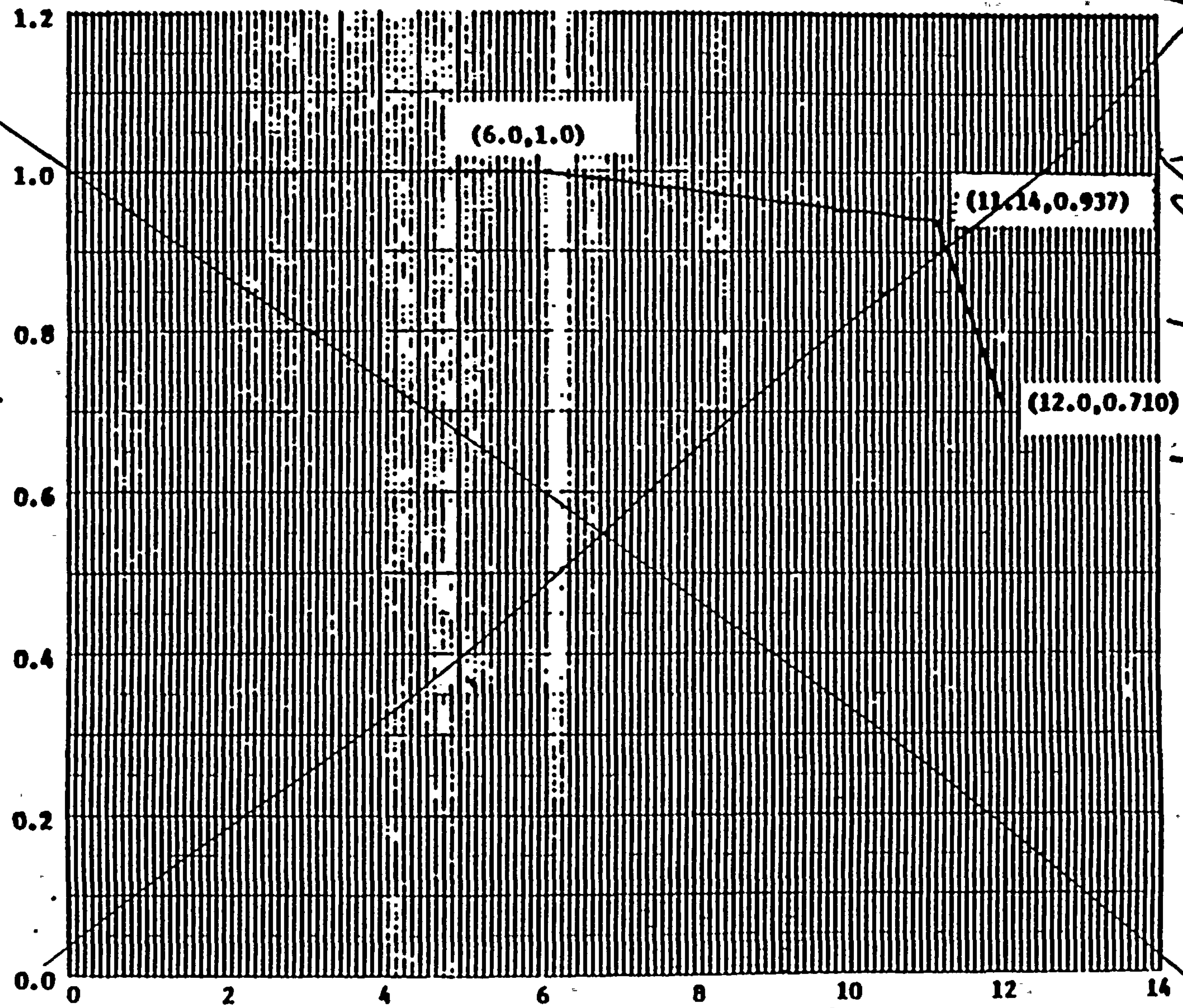

FIGURE 3.2-2(a) D. C. COOK UNIT 2, K(Z) - NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT FOR EXXON NUCLEAR CO. FUEL

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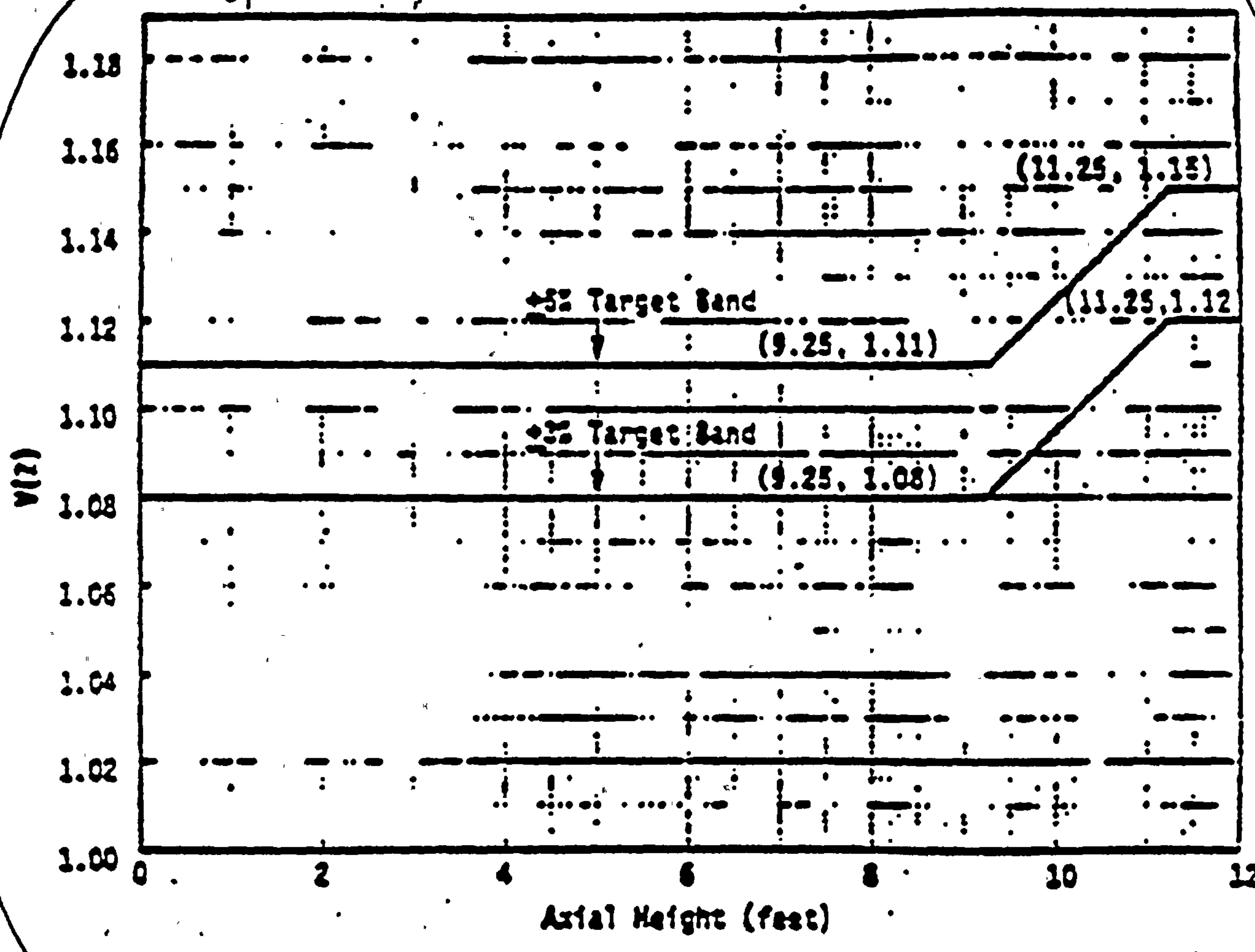

Figure 3.2-3 V(Z) As A Function of Core Height

NUCLEAR ENTHALPY HOT CHANNEL FACTOR - $F^N_{\Delta H}$

LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall be limited by the following relationship~~s~~:

$\cancel{F^N_{\Delta H} \leq 1.48 \; [1 + 0.2 \; (1-P)]}$ ~~(for Westinghouse fuel)~~

~~and~~ $F^N_{\Delta H} \leq \cancel{1.49}\, F_{\Delta H}^{RTP} \; [1 + \cancel{0.2}\, PF_{\Delta H} \; (1-P)]$ ~~(For Exxon Nuclear Co. fuel)~~

where P is the fraction of RATED THERMAL POWER

APPLICABILITY: MODE 1

ACTION:

With $F^N_{\Delta H}$ exceeding its limit:

a. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours,

b. Demonstrate through in-core mapping that $F^N_{\Delta H}$ is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours, and

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION may proceed, provided that $F^N_{\Delta H}$ is demonstrated through in-core mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL POWER and within 24 hours after attaining 95% or greater RATED THERMAL POWER.

$F_{\Delta H}^{RTP}$ is The $F^N_{\Delta H}$ Limit at RATED THERMAL POWER specified in The Core Operating Limits Report (COLR)

$PF_{\Delta H}$ is The power factor multiplier for $F^N_{\Delta H}$ specified in The COLR

3/4.2.5 DNB and Tavg OPERATING PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB and Tavg related parameters shall be maintained within the following operational indicated limits:

a. DNB
   1. Reactor Coolant System Tavg, $\leq 578.7^{o}F^{*}$
   2. Pressurizer Pressure $\geq$ 2194 psig*/**
   3. Reactor Coolant System Total Flow $\geq$ 366,400 gpm***

b. Tavg
   1. Reactor Coolant System Tavg $\geq 542.8^{o}F^{*}$

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters shall be verified to be within their limits at least once per 12 hours.

4.2.5.2 The indicators used to determine RCS total flow shall be subjected to a CHANNEL CALIBRATION at least once per 18 months.

4.2.5.3 The RCS total flow shall be determined by a power balance around the steam generators at least once per 18 months.

4.2.5.4 The provisions of Specification 4.0.4 shall not apply to primary flow surveillances.

* Indicated average of at least three OPERABLE instrument loops.

** Limit not applicable during either a thermal power ramp in excess of 5% of RATED THERMAL POWER per minute or a thermal power step in excess of 10% of RTP.

*** Indicated value.

DNB PARAMETERS - MODE 1

See insert

LIMITING CONDITION FOR OPERATION

3.2.5.1 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-1:

a. Reactor Coolant System T_{avg}.

b. Pressurizer Pressure.

c. Reactor Coolant System Total Flow Rate.

APPLICABILITY: MODE 1

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1.1 Each of the parameters of Table 3.2-1 shall be verified to be within their limits at least once per 12 hours.

4.2.5.1.2 The RCS total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months.

4.2.5.1.3 The RCS total flow rate shall be determined by a power balance around the steam generators at least once per 18 months.

4.2.5.1.4 The provisions of Specification 4.0.4 shall not apply to primary flow surveillances.

TABLE 3.2-1

DNB PARAMETERS

Delete

PARAMETER	LIMITS 4 Loops in Operation
Reactor Coolant System T_{avg}	$\leq$ 576.3°F. (indicated)**
Pressurizer Pressure	$\geq$ 2205 psig*, **
Reactor Coolant System Total Flow Rate	$\geq 138.6 \times 10^6$ lbs/hr***

* Limit not applicable during either a THERMAL POWER ramp in excess of 5% RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% RATED THERMAL POWER.

** Indicated average of at least three OPERABLE instrument loops.

*** 3.5% penalty for measurement uncertainty included in this value.

POWER DISTRIBUTION LIMITS

DNB PARAMETERS - MODES 2, 3, 4 and 5

LIMITING CONDITION FOR OPERATION

3.2.5.2 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-2:

a. Reactor Coolant System T_{avg}.

b. Pressurizer Pressure.

Delete

APPLICABILITY: MODES 2, 3*, 4* and 5*

ACTION:

MODES 2 and 3*

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or open the reactor trip system breakers within the next hour.

MODES 4* and 5*

Within one hour either open the reactor trip system breakers or render the control rod drive system incapable of rod withdrawal.

SURVEILLANCE REQUIREMENTS

4.2.5.2 Each of the parameters of Table 3.2-2 shall be verified to be within their limits at least once per 12 hours.

* With the reactor trip system breakers in the closed position and the control rod drive system capable of rod withdrawal.

TABLE 3.2-2

DNB PARAMETERS

Delete

PARAMETER	LIMIT
Reactor Coolant System T_{avg}	$\leq$ 549.2°F. (Reactor Subcritical)*
Reactor Coolant System T_{avg}	$\leq$ 576.3°F. (Reactor Critical)*
Pressurizer Pressure	$\geq$ 2176 psig*

Reactor coolant loop operational requirements are contained in Specifications 3.4.1.1, 3.4.1.2.c and 3.4.1.3.c.

* Indicated average of at least three OPERABLE instrument loops.

ALLOWABLE POWER LEVEL - APL

LIMITING CONDITION FOR OPERATION

3.2.6 THERMAL POWER shall be less than or equal to ALLOWABLE POWER LEVEL (APL), given by the following relationships:

~~Westinghouse Fuel~~

$$APL = \text{min over Z of } \frac{\cancel{1.97}\ F_Q^{RTP}\ K(Z)}{F_Q(Z) \times V(Z) \times F_p} \times 100\%, \text{ or } 100\%, \text{ whichever is less.}$$

~~Exxon Nuclear Co. Fuel~~

$$\cancel{APL = \text{min over Z of } \frac{2.10\ K(Z)}{F_Q(Z) \times V(Z) \times F_p} \times 100\%, \text{ or } 100\%, \text{ whichever is less.}}$$

- $F_Q(Z)$ is the measured hot channel factor, including a 3% manufacturing tolerance uncertainty and a 5% measurement uncertainty.

- V(Z) is the function defined in The COLR. ~~Figure 3.2-3 which corresponds to the target band.~~

- F_p = 1.00 except when successive steady-state power distribution maps indicate an increase in MAX OVER Z OF $\frac{F_Q(Z)}{K(Z)}$ ~~peak pin power, $F_{\Delta H}$~~, with exposure. Then either of the following penalties, F_p, shall be taken:

  F_p = 1.02 or,

  F_p = 1.00 provided that Surveillance Requirement 4.2.6.2 is satisfied once per 7 Effective Full Power Days until 2 successive maps indicate that the ~~peak pin $F_{\Delta H}$~~ MAX OVER Z OF $F_Q(Z)/K(Z)$ is not increasing.

- The above limit is not applicable in the following core regions.

  1) Lower core region 0% to 10% inclusive.
  2) Upper core region 90% to 100% inclusive.

APPLICABILITY: MODE 1

- F_Q^{RTP} is the F_Q limit at RATED THERMAL POWER specified in the Core Operating Limits Report (COLR)

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	NOT APPLICABLE
2. Power Range, Neutron Flux	≤ 0.5 seconds*
3. Power Range, Neutron Flux, High Positive Rate	NOT APPLICABLE
4. Power Range, Neutron Flux, High Negative Rate	≤ 0.5 seconds*
5. Intermediate Range, Neutron Flux	NOT APPLICABLE
6. Source Range, Neutron Flux	NOT APPLICABLE
7. Overtemperature ΔT	≤ 6.0 seconds*
8. Overpower ΔT	NOT APPLICABLE
9. Pressurizer Pressure--Low	≤ ~~1.0~~ 2.0 seconds
10. Pressurizer Pressure--High	≤ ~~1.0~~ 2.0 seconds
11. Pressurizer Water Level--High	~~NOT APPLICABLE~~ ≤ 2.0 SECONDS

*Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

TABLE 3.3-2 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
12. Loss of Flow - Single Loop (Above P-8)	≤ ~~0.6~~ 1.0 seconds
13. Loss of Flow - Two Loops (Above P-7 and below P-8)	≤ ~~0.6~~ 1.0 seconds
14. Steam Generator Water Level--Low-Low	≤ ~~1.5~~ 2.0 seconds
15. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	NOT APPLICABLE
16. Undervoltage-Reactor Coolant Pumps	≤ ~~1.2~~ 1.5 seconds
17. Underfrequency-Reactor Coolant Pumps	≤ 0.6 seconds
18. Turbine Trip	
A. Low Fluid Oil Pressure	NOT APPLICABLE
B. Turbine Stop Valve	NOT APPLICABLE
19. Safety Injection Input from ESF	NOT APPLICABLE
20. Reactor Coolant Pump Breaker Position Trip	NOT APPLICABLE

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
4. STEAM LINE ISOLATION		
a. Manual	Not Applicable	Not Applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure--High-High	≤ 2.9 psig	≤ 3.0 psig
d. Steam Flow in Two Steam Lines--High Coincident with T_{avg}--Low-Low	≤ A function defined as follows: A Δp corresponding to ~~1.47~~ 1.6 x 10^6 lbs/hr steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to ~~1.02~~ 4.5 x 10^6 lbs/hr at full load. $T_{avg} \geq 541^{\circ}$ F.	≤ A function defined as follows: A Δp corresponding to ~~1.62~~ 4.78 x 10^6 lbs/hr steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to ~~1.07~~ 4.55 x 10^6 lbs/hr at full load. $T_{avg} \geq 539^{\circ}$ F.
e. Steam Line Pressure--Low	≥ 600 psig steam line pressure	≥ 585 psig steam line pressure
5. TURBINE TRIP AND FEEDWATER ISOLATION		
a. Steam Generator Water Level--High-High	< 67% of narrow range instrument span each steam generator	< 68% of narrow range instrument span each steam generator

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual	
a. Safety Injection (ECCS)	Not Applicable
Feedwater Isolation	Not Applicable
Reactor Trip (SI)	Not Applicable
Containment Isolation-Phase "A"	Not Applicable
Containment Purge and Exhaust Isolation	Not Applicable
Auxiliary Feedwater Pumps	Not Applicable
Essential Service Water System	Not Applicable
Containment Air Recirculation Fan	Not Applicable
b. Containment Spray	Not Applicable
Containment Isolation-Phase "B"	Not Applicable
Containment Purge and Exhaust Isolation	Not Applicable
c. Containment Isolation-Phase "A"	Not Applicable
Containment Purge and Exhaust Isolation	Not Applicable
d. Steam Line Isolation	Not Applicable
2. Containment Pressure-High	
a. Safety Injection (ECCS)	~~Not Applicable~~ ≤ 27.0*
b. Reactor Trip (from SI)	~~Not Applicable~~ ≤ 3.0
c. Feedwater Isolation	~~Not Applicable~~ ≤ 8.6
d. Containment Isolation-Phase "A"	Not Applicable
e. Containment Purge and Exhaust Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Essential Service Water System	Not Applicable

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	≤ 24.0*/12.0#
b. Reactor Trip (from SI)	≤ 2.0
c. Feedwater Isolation	≤ 8.0
d. Containment Isolation-Phase "A"	≤ 18.0#
e. Containment Purge and Exhaust Isolation	Not Applicable
f. Motor Driven Auxiliary Feedwater Pumps	≤ 60.0
g. Essential Service Water System	≤ 48.0*/13.0#
4. Differential Pressure Between Steam Lines-High	
a. Safety Injection (ECCS)	≤ 12.0#/24.0##
b. Reactor Trip (from SI)	≤ 2.0
c. Feedwater Isolation	≤ 8.0
d. Containment Isolation-Phase "A"	≤ 18.0#/28.0##
e. Containment Purge and Exhaust Isolation	Not Applicable
f. Motor Driven Auxiliary Feedwater Pumps	≤ 60.0
g. Essential Service Water System	≤ 13.0#/48.0##
5. Steam Flow in Two Steam Lines - High Coincident with T_{avg}--Low-Low	
a. Safety Injection (ECCS)	Not Applicable
b. Reactor Trip (from SI)	Not Applicable
c. Feedwater Isolation	Not Applicable
d. Containment Isolation-Phase "A"	Not Applicable
e. Containment Purge and Exhaust Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Essential Service Water System	Not Applicable
h. Steam Line Isolation	≤ ~~10~~ 13

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION		RESPONSE TIME IN SECONDS
6. Steam Line Pressure--Low		
a. Safety Injection (ECCS)		≤ 12.0*/24.0**
b. Reactor Trip (from SI)		≤ 2.0
c. Feedwater Isolation		≤ 8.0
d. Containment Isolation-Phase "A"		≤ 18.0*/28.0**
e. Containment Purge and Exhaust Isolation		Not Applicable
f. Motor Driven Auxiliary Feedwater Pumps		≤ 60.0
g. Essential Service Water System		≤ 14.0*/48.0**
h. Steam Line Isolation		≤ 8.0
7. Containment Pressure--High-High		
a. Containment Spray		≤ 45.0
b. Containment Isolation-Phase "B"		~~Not Applicable~~
c. Steam Line Isolation		~~≤ 7.0~~ 11.0
d. Containment Air Recirculation Fan		≤ ~~600.0~~
8. Steam Generator Water Level--High-High		
a. Turbine Trip	≤ 2.5	~~Not Applicable~~
b. Feedwater Isolation	≤ 11.0	~~Not Applicable~~
9. Steam Generator Water Level--Low-Low		
a. Motor Driven Auxiliary Feedwater Pumps		≤ 60.0
b. Turbine Driven Auxiliary Feedwater Pump(s)		≤ 60.0
10. 4160 volt Emergency Bus Loss of Voltage		
a. Motor Driven Auxiliary Feedwater Pumps		≤ 60.0
11. Loss of Main Feedwater Pumps		
a. Motor Driven Auxiliary Feedwater Pumps		≤ 60.0
12. Reactor Coolant Pump Bus Undervoltage		
a. Turbine Driven Auxiliary Feedwater Pump(s)		≤ 60.0

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 a. The reactor coolant loops listed below shall be OPERABLE and in operation as required by items b, c, and d:

1. Reactor Coolant Loop 1 and its associated steam generator and reactor coolant pump,
2. Reactor Coolant Loop 2 and its associated steam generator and reactor coolant pump,
3. Reactor Coolant Loop 3 and its associated steam generator and reactor coolant pump,
4. Reactor Coolant Loop 4 and its associated steam generator and reactor coolant pump.

b. At least two of the above coolant loops shall be OPERABLE and at least one loop in operation if the reactor trip breakers are in the open position, or the control rod drive system is not capable of rod withdrawal.*

c. At least ~~three~~ TWO of the above coolant loops shall be OPERABLE and in operation when the reactor trip system breakers are in the closed position and the control rod drive system is capable of rod withdrawal.

d. At least three of the above coolant loops shall be OPERABLE and in operation above P-12. (Refer to Technical Specification 3.3.2.1, Table 3.3-3 for instrumentation requirements.)

APPLICABILITY: MODE 3

* All reactor coolant pumps may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration**, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

** For purposes of this specification, addition of water from the RWST does not constitute a dilution activity provided the boron concentration in the RWST is greater than or equal to the minimum required by specification 3.1.2.8.b.2.

LIMITING CONDITION FOR OPERATION

3.4.1.3 a. The coolant loops listed below shall be OPERABLE and in operation as required by items b and c:

1. Reactor Coolant Loop 1 and its associated steam generator and reactor coolant pump,*

2. Reactor Coolant Loop 2 and its associated steam generator and reactor coolant pump,*

3. Reactor Coolant Loop 3 and its associated steam generator and reactor coolant pump,*

4. Reactor Coolant Loop 4 and its associated steam generator and reactor coolant pump,*

5. Residual Heat Removal - East, **

6. Residual Heat Removal - West **

b. At least two of the above coolant loops shall be OPERABLE and at least one loop in operation if the reactor trip breakers are in the open position, or the control rod drive system is not capable of rod withdrawal.***

c. At least TWO ~~three~~ of the above reactor coolant loops shall be OPERABLE and in operation when the reactor trip system breakers are in the closed position and the control rod drive system is capable of rod withdrawal.

APPLICABILITY: MODES 4 and 5

* A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures less than or equal to 152°F unless 1) the pressurizer water volume is less than 62% of span or 2) the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures. Operability of a reactor coolant loop(s) does not require an OPERABLE auxiliary feedwater system.

** The normal or emergency power source may be inoperable in MODE 5.

*** All reactor coolant pumps and residual heat removal pumps may be de-energized for up to 1 hour provided 1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration****, and 2) core outlet temperature is maintained at least 10°F below saturation temperature.

**** For purposes of this specification, addition of water from the RWST does not constitute a dilution activity provided the boron concentration in the RWST is greater than or equal to the minimum required by specification 3.1.2.8.b.2 (MODE 4) or 3.1.2.7.b.2 (MODE 5).

REACTOR COOLANT SYSTEM

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4 The pressurizer shall be OPERABLE with a water volume less than or equal to ~~62%~~ 92% of span and at least 150 kW of pressurizer heaters.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With the pressurizer inoperable due to an inoperable emergency power supply to the pressurizer heaters, either restore the inoperable emergency power supply within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours. With the pressurizer otherwise inoperable, be in at least HOT SHUTDOWN with the reactor trip breakers open within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.4.1 The pressurizer water volume shall be determined to be within its limit at least once per 12 hours.

4.4.4.2 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by transferring power from the normal to the emergency power supply and energizing the required capacity of heaters.

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 1 GPM total primary-to-secondary leakage through all steam generators and 500 gallons per day through any one steam generator,

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. ~~52 GPM~~ CONTROLLED LEAKAGE CORRESPONDING TO A SEAL LINE RESISTANCE GREATER THAN OR EQUAL TO 0.2268 ft/gpm².

f. 1 GPM leakage from any reactor coolant system pressure isolation valve specified in Table 3.4-0.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any reactor coolant system pressure isolation valve(s) leakage greater than the above limit, except when:

1. The leakage is less than or equal to 5.0 gpm, and

2. The most recent measured leakage does not exceed the previous measured leakage* by an amount that reduces the

*To satisfy ALARA requirements, measured leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

margin between the most recent measured leakage and the maximum limit of 5.0 gpm by 50% or more,

declare the leaking valve* inoperable and isolate the high pressure portion of the affected system from the low pressure portion by the use of at least two closed valves, one of which may be the OPERABLE check valve and the other a closed de-energized motor operated valve. Verify the isolated condition of the closed de-energized motor operated valve at least once per 24 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere particulate radioactivity monitor at least once per 12 hours.

b. Monitoring the containment sump inventory and discharge at least once per 12 hours.

c. Monitoring the CONTROLLED LEAKAGE to the, reactor coolant pump seals at least once per 31 days, insert K

d. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours during steady state operation, and

e. Monitoring the reactor head flange leakoff system at least once per 24 hours.

4.4.6.2.2 Each reactor coolant system pressure isolation valve specified in Table 3.4-0 shall be demonstrated OPERABLE pursuant to Specification 4.0.5, except that in lieu of any leakage testing required by Specification 4.0.5, each valve shall be demonstrated OPERABLE by verifying leakage to be within its limit prior to entering MODE 3:

a. After each refueling outage;

b. Whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months;

* No Report required (6.9.1) unless the valve has been declared inoperable.

INSERT K

The seal line resistance is equal to 2.31*Pd/Q^2, where Pd is the charging pump discharge pressure minus the RCS pressure in psi, and Q is the CONTROLLED LEAKAGE in gpm.

ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each reactor coolant system accumulator shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water volume of between ~~929~~ 921 and 971 cubic feet,

c. A boron concentration between 2400 ppm and 2600 ppm, and

d. A nitrogen cover-pressure of between ~~599~~ 585 and ~~644~~ 658 psig.

APPLICABILITY: MODES 1, 2 and 3*.

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within one hour or be in HOT SHUTDOWN within the next 12 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in HOT STANDBY within one hour and be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying the contained borated water volume and nitrogen cover-pressure in the tanks, and

2. Verifying that each accumulator isolation valve is open.

*Pressurizer Pressure above 1000 psig.

d. At least once per 18 months by:

1. Verifying automatic isolation and interlock action of the RHR system from the Reactor Coolant System when the Reactor Coolant System pressure is above 600 psig.*

2. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

e. At least once per 18 months, during shutdown, by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection test signal.

2. Verifying that each of the following pumps start automatically upon receipt of a safety injection test signal:

a) Centrifugal charging pump

b) Safety injection pump

c) Residual heat removal pump

f. By verifying that each of the following pumps develops the indicated differential ~~discharge~~ pressure on recirculation flow when tested pursuant to Specification 4.0.5:

1. Centrifugal charging pump ≥ ~~2405 psig~~ 2290 psid
2. Safety Injection pump ≥ ~~1445 psig~~ 1385 psid
3. Residual heat removal pump ≥ ~~195 psig~~ 160 psid

g. By verifying the correct position of each mechanical stop for the following Emergency Core Cooling System throttle valves:

1. Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE.

* The provisions of Specification 4.0.7 are applicable.

SURVEILLANCE REQUIREMENTS (Continued)

2. At least once per 18 months.

Boron Injection Throttle Valves	Safety Injection Throttle Valves
Valve Number	Valve Number
1. 2-SI-141 L1	1. 2-SI-121 N
2. 2-SI-141 L2	2. 2-SI-121 S
3. 2-SI-141 L3	
4. 2-SI-141 L4	

h. By performing a flow balance test during shutdown following completion of modifications to the ECCS subsystem that alter the subsystem flow characteristics and verifying the following flow rates:

Boron Injection System Single Pump*	Safety Injection System Single Pump**
Loop 1 Boron Injection Flow 117.5 gpm	Loop 1 and 4 Cold Leg Flow ≥ 300 gpm
Loop 2 Boron Injection Flow 117.5 gpm	Loop 2 and 3 Cold Leg Flow ≥ 300 gpm
Loop 3 Boron Injection Flow 117.5 gpm	**Combined Loop 1,2,3 and 4 Cold Leg Flow (single pump) ≤ 640 gpm. Total SIS (single pump) flow, including miniflow, shall not exceed 700 gpm.
Loop 4 Boron Injection Flow 117.5 gpm	

Insert h

~~*The flow rate in each Boron Injection (BI) line should be adjusted to provide 117.5 gpm (nominal) flow into each loop. Under these conditions there is zero mini-flow and 80 gpm simulated RCP seal injection line flow. The actual flow in each BI line may deviate from the nominal so long as the difference between the highest and lowest flow is 10 gpm or less and the total flow to the four branch lines does not exceed 470 gpm. Minimum flow (total flow) required is 345.8 gpm to the three most conservative (lowest flow) branch lines.~~

INSERT H

The flow rate in each Boron Injection (BI) line should be adjusted to provide 117.5 gpm (nominal) flow in each loop. Under these conditions there is zero mini-flow and 80 gpm, plus or minus 5 gpm, simulated RCP seal injection line flow. The actual flow rate in each BI line may deviate from the nominal so long as:

a) the difference between the highest and lowest flow rate is 25 gpm or less.

b) the total flow rate to the four branch lines does not exceed 470 gpm.

c) the minimum flow rate through the three most conservative (lowest flow) branch lines must not be less than 300 gpm.

d) the charging pump discharge resistance (2.31*Pd/Qd^2) must not be less than 4.73E-3 ft/gpm^2 and must not be greater than 9.27E-3 ft/gpm^2, (Pd is the pump discharge pressure at runout; Qd is the total pump flow rate.

EMERGENCY CORE COOLING SYSTEMS

REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.5 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A minimum contained volume of 350,000 gallons of borated water,

b. Between 2400 and 2600 ppm of boron, and

c. A minimum water temperature of ~~80~~ 70°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the refueling water storage tank inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the contained borated water volume in the tank, and

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature.

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition for increased load events occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of ~~2.0%~~ 1.6% $\Delta k/k$ is initially required to control the reactivity transient and automatic ESF is assumed to be available.

Technical Specification requirements call for verification that the SHUTDOWN MARGIN is greater than or equal to that which would be required for the MODE 3 low temperature value, 350°F, prior to blocking safety injection on either the P-11 or P-12 permissive interlocks. This assures in the event of an inadvertent opening of two cooldown steam dump valves that adequate shutdown reactivity is available to allow the operator to identify and terminate the event.

With T_{avg} < 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 1% $\Delta k/k$ SHUTDOWN MARGIN provides adequate protection for this event.

In shutdown MODES 4 and 5 when heat removal is provided by the residual heat removal system, active reactor coolant system volume may be reduced. Increased SHUTDOWN MARGIN requirements when operating under these conditions is provided for high reactor coolant system boron concentrations to ensure sufficient time for operator response in the event of a boron dilution transient.

The SHUTDOWN MARGIN requirements are based upon the limiting conditions described above and are consistent with FSAR safety analysis assumptions.

3/4.1.1.3 BORON DILUTION

A minimum flow rate of at least 3000 GPM provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual during boron concentration reductions in the Reactor Coolant System. A flow rate of at least 3000 GPM will circulate an equivalent Reactor Coolant System volume of 12,612 cubic feet in approximately 30 minutes. The reactivity change rate associated with boron reductions will therefore be within the capability for operator recognition and control.

BASES

With the RCS average temperature above 200°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps and safety injection pumps, except the required OPERABLE charging pump, to be inoperable below 152°F, unless the reactor vessel head is removed, provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The boration capability of either system is sufficient to provide the required SHUTDOWN MARGIN from expected operating conditions after xenon decay and cooldown to 200°F. The maximum expected boration capability usable volume
7715
requirement is 3700 gallons of 20,000 ppm borated water from the boric acid storage tanks or ~~118,000~~ gallons of borated water from the refueling water
160,122
storage tank. The required RWST volume is based on an assumed boron concentration of 2000 ppm. The minimum RWST boron concentration required by
2400
the post-LOCA long-term cooling analysis is 2400 ppm. The minimum contained RWST volume is based on ECCS considerations. See Section B 3/4.5.5. ~~The boration source volume from the boric acid storage tank has conservatively been increased to 5650 gallons. This value was chosen to be consistent with Unit 1.~~

With the RCS temperature below 200°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

The boron capability required below 200°F is sufficient to provide the required MODE 5 SHUTDOWN MARGIN after xenon decay and cooldown from 200°F to 140°F. This condition requires usable volumes of either 4300 gallons of 20,000 ppm borated water from the boric acid storage tanks or 90,000 gallons of borated water from the refueling water storage tank. The value for the boric acid storage tank volume includes sufficient boric acid to borate to
2190
2000 ppm. The required RWST volume is based on an assumed boron concentration of 2000 ppm. The minimum RWST boron concentration required by the post-LOCA
2400
long-term cooling analysis is 2400 ppm.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.6 and 9.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The OPERABILITY of boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the calculated DNBR in the core at or above design during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

The limits on $F_Q(Z)$ and $F^N_{\Delta H}$ ARE SPECIFIED IN THE CORE OPERATING LIMITS REPORT (COLR) ~~for Westinghouse supplied fuel at a core average power of 3411 MWt are 1.97 and 1.48, respectively, which assure consistency with the allowable heat generation rates developed for a core average thermal power of 3391 MWt. The limits on $F_Q(Z)$ and $F^N_{\Delta H}$ for ENC supplied fuel have been established for a core thermal power of 3411 MWt. The limit on $F_Q(Z)$ is 2.10. The limit on $F^N_{\Delta H}$ is 1.49. The analyses supporting the Exxon Nuclear Company limits are valid for an average steam generator tube plugging of up to 10% and a maximum plugging of one or more steam generators of up to 15%. In establishing the limits, a plant system description with improved accuracy was employed during the reflood portion~~ of ~~the LOCA Transient. With respect to the Westinghouse supplied fuel, the minimum projected excess margin to ECCS limits will more than offset the impact of increased steam generator tube plugging.~~

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

The limits on AXIAL FLUX DIFFERENCE assure that the $F_Q(Z)$ upper bound envelope is not exceeded during either normal operation or in the event of xenon redistribution following power changes. The $F_Q(Z)$ upper bound envelope is ~~1.97 times the average fuel rod heat flux for Westinghouse supplied fuel and 2.10 times the average fuel rod heat flux for Exxon Nuclear Company supplied fuel~~. SPECIFIED IN THE CORE OPERATING LIMITS REPORT (COLR)

Target flux difference is determined at equilibrium xenon conditions. The full length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady state operation at high power levels. The value of the

POWER DISTRIBUTION LIMITS

BASES

target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

Although it is intended that the plant will be operated with the AXIAL FLUX DIFFERENCE within the target band about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1 hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits ~~of Figure 3.2-1~~ while at THERMAL POWER levels above 50% of RATED THERMAL POWER. For THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

Specified in the CORE OPERATING LIMITS REPORT

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message if the AFD for at least 2 of 4 or 2 of 3 OPERABLE excore channels are outside the target band and the THERMAL POWER is greater than 90% or 0.9 x APL of RATED THERMAL POWER (whichever is less). During operation at THERMAL POWER levels between 50% and 90% or 0.9 x APL of RATED THERMAL POWER (whichever is less) and between 15% and 50% RATED THERMAL POWER, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour and 2 hours, respectively.

~~Figure B 3/4 2-1 shows a typical monthly target band.~~ delete

The basis and methodology for establishing these limits is presented in topical report ~~XN-NF-77-57~~, "~~Exxon Nuclear Power Distribution Control for PWRs - Phase II" and Supplements 1 and 2 to that report.~~

WCAP-8385, "Power Distribution Control and Load Following Procedures."

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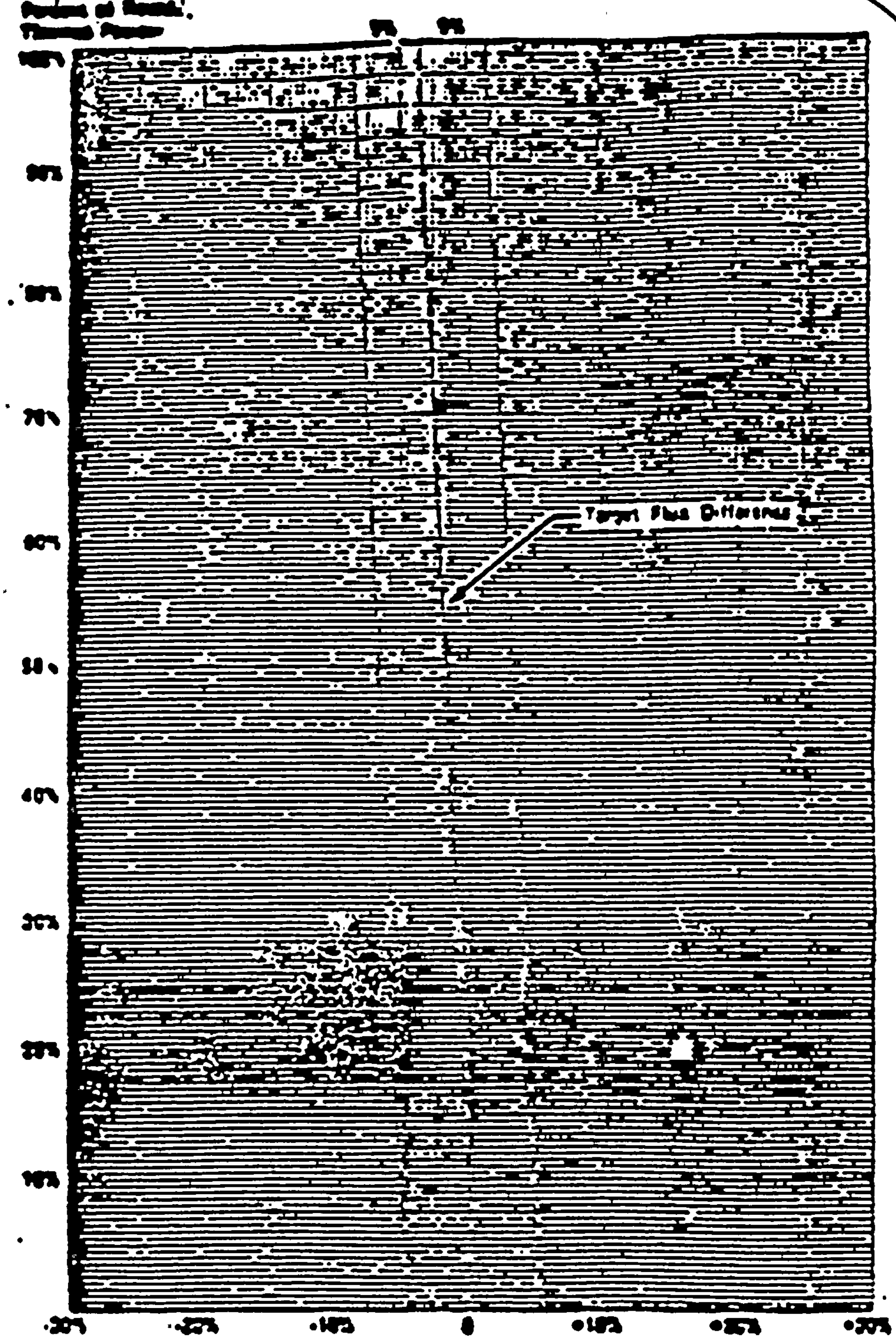

Figure B 3/4 2-1 TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR, AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor and nuclear enthalpy rise hot channel factor ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2.1, 4.2.2.2, 4.2.3, 4.2.6.1 and 4.2.6.2. This periodic surveillance is sufficient to ensure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps from the group demand position.

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6.

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained.

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

AS specified in the Core Operating Limits Report (COLR)

$F^N_{\Delta H}$ will be maintained ~~within its limits~~ provided conditions a. through d. above are maintained. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits. The form of this relaxation for DNBR limits is discussed in Section 2.1.1 of the basis.

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. 5% is the appropriate allowance for a full core map taken with the incore detector flux mapping system and 3% is the appropriate allowance for manufacturing tolerance.

~~A measurement error allowance of 4% on $F^N_{\Delta H}$ has been included in the Technical Specification limit so that measured values may be compared directly to the $F^N_{\Delta H}$ limit.~~

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BASES: (Continued)

Figure B 3/4 2-2 illustrates the implementation of the limits as a function of power. A measured flow will result in a limiting value for R which must be obtained from Figure 3.2-4 or Figure 3.2-5. From this limiting R, a limiting $F^N_{\Delta H}$ can be obtained because:

Westinghouse Fuel

$F^N_{\Delta H} = 1.48 \times R \times [1.0+0.2(1.0-P)]$,

Exxon Nuclear Company Fuel

$F^N_{\Delta H} = 1.49 \times R \times [1.0+0.2(1.0-P)]$

$$\text{Where: } P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

Figure B 3/4 2-2 displays two limiting DNBR $F^N_{\Delta H}$ curves for Exxon Nuclear Company fuel for flows of 36.77 X 10^4 gpm, and 37.63 X 10^4 gpm. Also displayed on Figure B 3/4 2-2 is the limit on $F^N_{\Delta H}$ which results from the LOCA analysis for Exxon Nuclear Company fuel. $F^N_{\Delta H}$ must be maintained below and to the left of both the applicable DNBR $F^N_{\Delta H}$ limit and the LOCA $F^N_{\Delta H}$ limit.

For Westinghouse fuel there is only one $F^N_{\Delta H}$ limit. It must be obtained from the applicable relationships among R, $F^N_{\Delta H}$, P, and flow.

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. 5% is the appropriate allowance for a full core map taken with the incore detector flux mapping system and 3% is the appropriate allowance for manufacturing tolerance.

When RCS flow rate and $F^N_{\Delta H}$ are measured, no additional allowances are necessary prior to comparison with the limits of Specification 3.2.3. Measurement errors of ~~3.5%~~ 2.1% for RCS flow total flow rate and 4% for $F^N_{\Delta H}$ have been allowed for in determination of the design DNBR value and in the determination of the LOCA/ECCS limit.

Margin between the safety analysis limit DNBRs (1.69 and 1.61 for the Vantage 5 typical and thimble cells, respectively and 1.43 and 1.40 for the ANF fuel typical and thimble cells) and the design limit DNBRs (1.23 and 1.22 for the Vantage 5 typical and thimble cells, and 1.39 and 1.36 for the ANF fuel typical and thimble cells) is maintained. A fraction of this margin is utilized to accommodate applicable transition core penalties and the appropriate fuel rod bow DNBR penalty for the Vantage 5 fuel (equal to 1.3% per WCAP-8691, Rev. 1). The remainder of the margin between design and safety analysis DNBR limits can be used for plant design flexibility.

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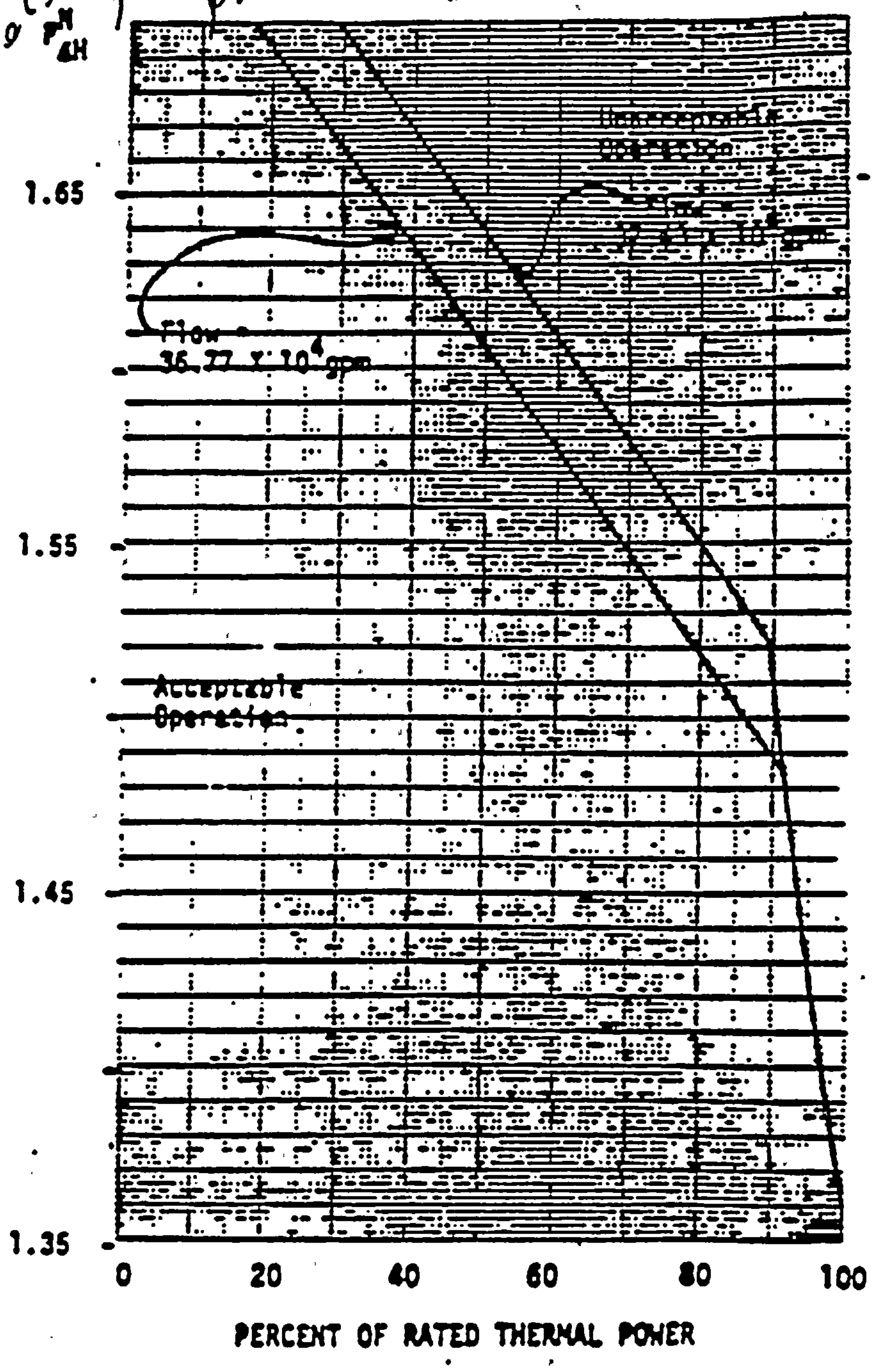

FIGURE B 3/4 2-2 ILLUSTRATIVE EXAMPLE OF $F^N_{\Delta H}$ LIMIT VERSUS PERCENT THERMAL POWER FOR EXXON FUEL

3/4.2.4 QUADRANT POWER TILT RATIO

The quadrant power tilt ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02 at which corrective action is required provides DNB and linear heat generation rate protection with x-y plane power tilts.

The two hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the power by 3 percent from RATED THERMAL POWER for each percent of tilt in excess of 1.0.

3/4.2.5 DNB PARAMETERS

see insert E

~~The limits on the DNB related parameters in MODE 1 assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the assumptions of the safety analysis and have been analytically demonstrated adequate to maintain design DNBR throughout each analyzed transient. The indicated values of T_{avg}, pressurizer pressure, and flow include allowances for instrument errors.~~

The 12 hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 12-hour surveillance of the RCS flow measurement is adequate to detect flow degradation. The CHANNEL CALIBRATION performed after refueling ensures the accuracy of the shiftly flow measurement. The total flow is measured after each refueling based on a secondary side calorimetric and measurements of primary loop temperatures.

The limits on pressurizer pressure and T_{avg} in MODES 2 and 3 provide protection against DNB resulting from an uncontrolled rod withdrawal from a subcritical condition. The indicated values of T_{avg} and pressurizer pressure include allowances for instrument errors.

3/4.2.6 ALLOWABLE POWER LEVEL - APL

see insert

~~The power distribution control procedure, PDC-II, manages core power distributions such that Technical Specification limits on $F_Q(Z)$ are not violated during normal operation and limits on MDNBR are not violated during steady-state, load-follow, and anticipated transients. The V(Z) factor given in the Technical Specifications provides the means for predicting the maximum $F_Q(Z)$ distribution anticipated during operation under the PDC-II procedure taking into account the incore measured equilibrium power distribution. A comparison of the maximum $F_Q(Z)$ with the Technical Specification limit determines the power level (APL) below which the Technical Specification limit can be protected by PDC-II. This comparison is done by calculating APL, as defined in specification 3.2.6.~~

INSERT E to page B 3/4 2-5

The limits on the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The Tavg ≤ 578.7 Degrees F and Pressurizer Pressure ≥ 2194 psig (indicated value) are consistent with the UFSAR assumptions and have been analytically demonstrated adequate to maintain the core at or above the design DNBR throughout each analyzed transient with allowance for measurement uncertainty. The Tavg ≥ 542.8 Degrees F is consistent with a safety analysis performed to demonstrate that the plant may operate on a linear control program where the analytical limit of Tavg at 100% RATED THERMAL POWER may range from 541.4 Degrees F to 580.1 Degrees F. The core may be operated with indicated vessel average temperature at any value between the upper and lower limits. ~~Pressurizer pressure is limited to two discrete nominal setpoints, with the lower limit of the indicated value of each setpoint set forth in the specifications.~~ The limits are consistent with the UFSAR assumptions and have been analytically demonstrated to be adequate to maintain the core at or above the applicable design limit DNBR values for each fuel type (which are listed in the bases for Section 2.1.1) throughout each analyzed transient.

Insert for B 3/4.2.6

Constant Axial Offset Control (CAOC) operation manages core power distributions such that Technical Specification limits on $F_Q(Z)$ are not violated during normal operation and limits on MDNBR are not violated during steady-state, load-follow, and anticipated transients. The V(Z) factor given in the Peaking Factor Limit Report and applied by the Technical Specifications provides the means for predicting the maximum $F_Q(Z)$ distribution anticipated during operation using CAOC taking into account the incore measured equilibrium power distribution. A comparison of the maximum $F_Q(Z)$ with the Technical Specification limit determines the power level (APL) below which the Technical Specification limit can be protected by CAOC. This comparison is done by calculating APL, as defined in specification 3.2.6.

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the RWST as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits of RWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.6 and 9.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The ECCS analyses to determine F_Q limits in Specifications 3.2.2 and 3.2.6 assumed a RWST water temperature of ~~80°F~~. This temperature value of the RWST water determines that of the spray water initially delivered to the containment following LOCA. It is one of the factors which determines the containment back-pressure in the ECCS analyses, performed in accordance with the provisions of 10 CFR 50.46 and Appendix K to 10 CFR 50.

70°F

3/4.7 PLANT SYSTEMS

BASES

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% of its design pressure of 1085 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. ~~The total relieving capacity for all valves on all of the steam lines is 17,153,800 lbs/hr which is 117 percent of the total secondary steam flow of 14,674,000 lbs/hr at 100% RATED THERMAL POWER.~~ insert A minimum of 2 OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Range Neutron Flux channels. The reactor trip setpoint reductions are derived on the following bases:

For 4 loop operation

$$SP = \frac{(X) - (Y)(V)}{X} \times (109)$$

Where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER

V = maximum number of inoperable safety valves per steam line

X = total relieving capacity of all safety valves per steam line in lbs./hours = 4,288,450

Y = maximum relieving capacity of any one safety valve in lbs./hour = 857,690

109 = Power Range Neutron Flux-High Trip Setpoint for 4 loop operation

The total rated relieving capacity of all valves on all of the steam lines is 17,153,800 lbs/hr which is at least 105% of the maximum secondary steam flow rate at 100% RATED THERMAL POWER.

3/4.9 REFUELING OPERATIONS

BASES

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: 1) the reactor will remain subcritical during CORE ALTERATIONS, and 2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the accident analyses. The value of 0.95 or less for K_{eff} includes a 1 percent delta k/k conservative allowance for uncertainties. Similarly, the boron concentration value of ~~2000~~ 2400 ppm or greater includes a conservative uncertainty allowance of 50 ~~ppm boron. The boron concentration requirement of specification 3.9.1.b has been conservatively increased to 2400 ppm to agree with the minimum concentration of the RWST.~~

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment building penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS.

The radioactive effluent release report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed member of the public from reactor releases and other nearby uranium fuel cycle sources (including doses from primary effluent pathways and direct radiation) for the previous 12 consecutive months to show conformance with 40 CFR 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1.

The radioactive effluent release report shall include the following information for each type of solid waste shipped offsite during the report period:

a. Volume (cubic meters),

b. Total curie quantity (specify whether determined by measurement or estimate),

c. Principal radionuclides (specify whether determined by measurement or estimate),

d. Type of waste (e.g., spent resin, compacted dry waste, evaporator bottoms),

e. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and

f. Solidifcation agent (e.g., cement).

The radioactive effluent release report shall include unplanned releases from the site to unrestricted areas of radioactive materials in gaseous and liquid effluent on a quarterly basis.

The radioactive effluent release reports shall include any change to the PROCESS CONTROL PROGRAM (PCP) and the OFFSITE DOSE CALCULATION MANUAL (ODCM) made during the reporting period.

MONTHLY REACTOR OPERATING REPORT

6.9.1.10 Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Director, Office Of Management and Program Analysis, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Office no later than the 15th of each month following the calendar month covered by the report.

CORE OPERATING LIMITS REPORT

6.9.1.11 [See Attached Section 6.9.1.11]

This is to be added to Section 6.9 - REPORTING REQUIREMENTS:

CORE OPERATING LIMITS REPORT

6.9.1.11 ~~x.x~~a Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

1. Moderator Temperature Coefficient BOL and EOL limits and 300 ppm surveillance limit for Specification 3/4.1.1.~~3~~4

2. Shutdown Bank Insertion Limit for Specification 3/4.1.3.5,

3. Control Bank Insertion Limits for Specification 3/4.1.3.6,

4. Axial Flux Difference limits, and target band, ~~and APL^ND~~ for Specification 3/4.2.1,

5. Heat Flux Hot Channel Factor, K(Z), ~~and~~ And ~~Power Factor Multiplier~~, ~~F_{xy}^{RTP}~~, V(Z) ~~W(Z)~~, ~~APL^ND~~ and ~~W(Z)~~$_{BL}$ for Specification 3/4.2.2 and 3/4.2.6,

6. and Nuclear Enthalpy Rise Hot Channel Factor and Power Factor Multiplier, ~~$R/R_1/R_2$ limits, and Rod Bow Penalty~~ for Specification 3/4.2.3, ~~and~~

~~7. Boron Concentration for Specification 3/4.9.1.~~

~~6.9.x.x.b~~ The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by NRC in:

1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY", July 1985 (W Proprietary).
   (Methodology for Specifications 3.1.1.~~3~~4 Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limit, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor, and ~~9.1 - Boron Concentration~~.)

2a. WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES - TOPICAL REPORT", September 1974 (W Proprietary).
   (Methodology for Specification 3.2.1 - Axial Flux Difference (Constant Axial Offset Control).)

2b. T. M. Anderson to K. Kniel (Chief of Core Performance Branch, NRC) January 31, 1980 -- Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package.
   (Methodology for Specification 3.2.1 - Axial Flux Difference (Constant Axial Offset Control).)

CORE OPERATING LIMITS REPORT (continued)

2c. NUREG-0800, Standard Review Plan, U. S. Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981. Branch Technical Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev.2, July 1981.
(Methodology for Specification 3.2.1 - Axial Flux Difference (Constant Axial Offset Control).)

delete

~~3. WCAP-10216-P-A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL FQ SURVEILLANCE TECHNICAL SPECIFICATION", June 1983 (W Proprietary).~~
~~(Methodology for Specifications 3.2.1 - Axial Flux Difference (Relaxed Axial Offset Control) and 3.2.2 - Heat Flux Hot Channel Factor (W(Z) surveillance requirements for F_Q Methodology).)~~

~~4a. WCAP-9220-P-A, Rev. 1, "WESTINGHOUSE ECCS EVALUATION MODEL-1981 VERSION", February 1982 (W Proprietary).~~
~~(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)~~

~~4b. WCAP-9561-P-A, ADD. 3, Rev.1, "BART A-1: A COMPUTER CODE FOR THE BEST ESTIMATE ANALYSIS OF REFLOOD TRANSIENTS - SPECIAL REPORT: THIMBLE MODELING W ECCS EVALUATION MODEL," July, 1986, (W Proprietary).~~
~~(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)~~

4c. WCAP-10266-P-A Rev.2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE", March 1987, (W Proprietary).
(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor).

[redacted] The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

[redacted] The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

ATTACHMENT 1 TO AEP:NRC:1071E

REASONS AND 10 CFR 50.92 ANALYSES FOR PROPOSED CHANGES TO THE DONALD C. COOK NUCLEAR PLANT UNITS 1 AND 2 TECHNICAL SPECIFICATIONS

I. Organization of the Submittal

The primary purpose of this submittal is to obtain approval to operate Unit 2 for Cycles 8 and 9 with a mixed core of Vantage 5 and ANF fuel. To facilitate the reviewer's task, we have organized the submission in several attachments listed below. One of these, Attachment 3, "Donald C. Cook Nuclear Plant Units 1 and 2, Summary of Proposed Technical Specifications Changes," identifies each proposed change by page and T/S number, briefly describes the change and the reason for the proposed change, and directs the reviewer to the appropriate supporting analysis or evaluation documentation and to the appropriate section of the 10 CFR 50.92 significant hazards consideration analysis.

The attachments are as follows:

This attachment, Attachment 1, contains a description of the organization of the submittal and our analysis of significant hazards considerations for the proposed T/S changes.

Attachment 2 contains the proposed T/S changes.

Attachment 3 contains a summary of the Unit 1 and Unit 2 proposed T/S changes with a description and the reasons for each proposed change.

Attachment 4 contains evaluations, T/S mark-ups, non-LOCA accident analyses, and LOCA analyses performed by Westinghouse in support of the Unit 2 Cycle 8 reload.

Attachment 5 contains the results of the calculation of mass and energy releases inside containment. This discussion has been taken from WCAP 11902, Supplement 1. Portions of this document which are needed to support operation of Unit 2 in Cycle 8 are included in this attachment.

Attachment 6 contains the justification for operation with pressurizer level increased from 67% to 92%. The material included in this attachment is part of a previous submittal to the NRC, WCAP 11902, "Pressurizer Operability Level Justification." WCAP 11902 was submitted on October 14, 1988, in AEP:NRC:1067. It is included with this submittal to facilitate the reviewer's task.

Attachment 7 is a copy of a sample COLR with expected Unit 2 Cycle 8 values.

Attachment 8 contains our analysis of significant hazards considerations for not analyzing seven events not in the Donald C. Cook Nuclear Plant Unit 2 licensing basis.

Attachment 9 is a copy of a letter from Joseph G. Giitter, NRC staff, of August 3, 1989, to Milton P. Alexich, AEPSC. This letter states that the seven events identified in Attachment 8 are not part of the Unit 2 licensing basis. The letter also requires a significant hazards consideration analysis of these seven events.

Attachment 10 is a copy of Part B of WCAP 10217-A. This WCAP is included as justification for a proposed T/S change in the power distribution limits section of the T/Ss. Westinghouse power distribution methodologies will be used in support of Unit 2 Cycle 8 in lieu of the ANF methodologies used in Cycles 4 through 7.

Attachment 11 is a copy of Section 1.2, "Major Analytical Assumptions," from WCAP 11908, "Containment Integrity Analysis for Donald C. Cook Nuclear Plant Units 1 and 2." WCAP 11908 was submitted on August 22, 1988 in AEP:NRC:1024D. Documentation of this analysis is provided to support our proposed residual heat removal pump surveillance requirement based on 10% degradation. It is included with this submittal to facilitate the reviewer's task.

II. Purpose of Proposed T/S Changes

The proposed T/S changes included in this submittal are intended to accomplish six purposes. In the following discussion, the proposed T/S changes are grouped according to these purposes. These groups are:

Group 1) Make changes that result from the analyses performed by our contractor, Westinghouse, to support operation of the Donald C. Cook Nuclear Plant, Unit 2 in Cycles 8 and 9 with a mixed core of V5 fuel and ANF fuel.

Group 2) Remove certain T/S requirements that are part of the Unit 2, Cycle 6, Amendment No. 82. These T/Ss all address concerns in transition modes of operation, Modes 3 and 4. They are not standard in the sense of the Standard Technical Specifications (STS). We believe those concerns can be safely addressed administratively as they are at other nuclear facilities.

Group 3) Make the Unit 2 T/Ss more nearly like the Unit 1 T/Ss. There is one change in this category. It is a proposal to increase the pressurizer water volume limit. This change was submitted for Unit 1 on October 14, 1988, with our submittal AEP:NRC:1067. It was approved in Amendment No. 126 to the Unit 1 operating license. For the convenience of the reviewer, the supporting evaluation is resubmitted in Attachment 6.

Group 4) Make administrative changes. Where substantive changes are proposed on a T/S page, some changes to enhance the clarity of the T/S are also proposed. In addition, some administrative changes result naturally from reformatting, page removals, and text movement. Administrative changes are proposed for both units.

Group 5) Modify a surveillance requirement for axial flux difference (AFD).

Group 6) Make a number of Unit 1 changes where the justifications are identical to or essentially identical to those for proposed Unit 2 changes. These proposed changes help to make the T/Ss for the two units more nearly alike.

Three of the Group 6 proposals correspond to proposed Unit 2 changes in Group (2) above, one to Group (5), and two to Group (1). The Group (1) changes are a proposal to change the discharge pressure requirement for the residual heat removal and safety injection pumps. Reference to the supporting analysis is included in Attachment 5.

III. <u>Overview of the Proposed T/S Changes and 10 CFR 50.92 Evaluations</u>

A summary of the proposed T/S and Bases changes has been included as Attachment 3 to this letter. Attachment 3 contains the following information for each proposed T/S change:

a) Reference to T/S page and section.

b) Reference to the Significant Hazards Consideration Analysis group.

c) Sequential identifier number.

d) Brief description.

e) Remarks which provide the reason for the change and a reference to safety analyses and evaluations as appropriate.

This attachment (Attachment 1) includes an overview of the proposed T/S changes and our 10 CFR 50.92 analysis for no significant hazards consideration.

We have grouped the proposed T/S changes in this discussion according to the six purposes described above.

1) T/S Change Group 1

Changes based on analyses performed to support Unit 2 Cycle 8 operation.

The changes in this category are based on Westinghouse analyses and evaluations performed using NRC approved methodologies. They are as follows:

a) Reactor Core Safety Limit Curves and OTdelta-T/OPdelta-T Reactor Trip Setpoints, T/S Figure 2.1-1 and T/S Table 2.2-1, Functional Units 7 and 8.

These changes are numbered 001, 007, 008, 011, 012, 013, 014, 015, 016, 019, 020, 021, 022, and 023 in Attachment 3. A new thermal design was performed for Cycle 8 operation. The results are described in Appendix B of Attachment 4, Section B.2.2.1. The proposed new safety limit curves are calculated for 3588 Mwt core power. They are conservatively applied to 3411 Mwt operation. The revised OTdelta-T/OPdelta-T setpoints which protect this thermal design are also discussed in Section B.2.2.1. The revised setpoints are provided in Attachment 4, Appendix A.

b) Design Flow, Footnote *, T/S Table 2.2-1

This change is numbered 002 in Attachment 3. Design flow is minimum measured flow (MMF) divided by four. MMF = 366,400 gpm. Design flow = 91,600. Analysis of DNB events with MMF using the Revised Thermal Design Procedure (RTDP) is discussed in Attachment 4, Appendix B, Section B.2.3. Table B.2-4 shows a loss of forced reactor coolant flow event to be a RTDP event. Use of the low flow trip is discussed in Attachment 4, Appendix B, Section B.3.5.1. The analysis value of the low flow trip is given in Attachment 4, Appendix B, Table B.2-2. Westinghouse has provided the design flow in Attachment 4, Appendix A.

c) Shutdown Margin (SDM), T/S 3/4.1.1.1 and 3/4.1.1.2

This change is numbered 026 in Attachment 3. The analyses performed for Cycle 8 assumed a SDM of 1.3% as discussed in Appendix B of Attachment 4, Section B.3.11.2. This value was also assumed for the mass and energy releases inside containment as discussed on Page S-3.3-12 of Attachment 5. The analysis of record for the mass and energy releases outside containment is discussed in Attachment 4, Section 5.4.2. This analysis assumed a SDM of 1.6%. Based on this limiting analysis, we propose to reduce the SDM for Unit 2 from the current 2.0% to 1.6%.

The SDM requirement of 2.0% resulted from an ANF analysis which will be superceded by the Westinghouse analysis in Cycle 8.

d) Borated Water Sources, T/S 3/4.1.2.8

This change is numbered 034 in Attachment 3. The new boric acid storage tank required volume calculated for Unit 2 is a bounding value which is expected to accommodate uprating to core thermal power of 3588 Mwt, fuel of increased enrichment for increased cycle length, and changes in vendor methodology. This change is indicated in Attachment 4, Table 6.1 and Attachment 4, Appendix A.

e) Rod Drop Time, T/S 3/4.1.3.4

This change is numbered 036 in Attachment 3. The intermediate flow mixer grid feature of the V5 fuel design increases the core pressure drop. Therefore, the control rod scram time to the dashpot has been increased to 2.7 seconds. This is discussed in Section 5.1.2 of Attachment 4 and in Section B.2.4 of Appendix B of Attachment 4.

f) Minimum Measured Flow, T/S 3/4.2.5

This change is numbered 043 in Attachment 3. A small change was made in the analysis assumption for primary flow. The analysis value for events not analyzed using the Revised Thermal Design Procedure (RTDP) is 354,000 gpm. The minimum measured flow (MMF) is 3.5% larger than this value conservatively allowing for measurement errors. The MMF is also the value used for flow in the DNB events analyzed with RTDP. This is discussed in Appendix B of Attachment 4, Section B.2.3.

g) Tavg Window, T/S 3/4.2.5

This change is numbered 041 in Attachment 3. Unit 2 has been analyzed for operation over a range of primary temperatures. This is reviewed in general in Appendix B of Attachment 4, Section B.2.1. The limiting temperature assumptions are addressed throughout the discussions of the transients and accidents when limiting assumptions are discussed. The values in T/S 3.2.5 are obtained from the analysis values shown in Table B.2-1 and Section B.2.1 of Attachment 4, Appendix B as follows:

	Lower Limit ^{o}F	High Limit ^{o}F
Analysis Value	547	576.0
Controller Allowance	-5.6	+4.1
Readability Allowance	+1.4	-1.4
T/S Limit	542.8	578.7

The analysis values are obtained from cases 2 and 3 of Table B.2-1 which apply to Cycles 8 and 9. The T/S limits are provided in Attachment 4, Appendix A.

Our proposed T/S upper temperature limit is 578.7^{o}F and our proposed T/S lower temperature limit is set conservatively at 543.9^{o}F. The lower limit is the value we plan to submit for Unit 1 in the future and was selected to make the T/Ss of the two units more consistent.

h) Minimum Pressurizer Pressure for Operation

This change is numbered 042 in Attachment 3. A new pressure allowance for control and readability was established for Unit 2. See Attachment 4, Appendix B, Section B.2.3, and Appendix C, Table C.3.1-1. SBLOCA used the same pressure assumptions shown for LBLOCA. The total allowance used in the analysis was:

Controller Allowance	35 psi
Readability Allowance	22 psi
Margin	6 psi
Total Allowance	63 psi

The T/S pressure limit is obtained analogously to obtaining the temperature limits above.

Analysis Value	2235 psig
Total Allowance	-63 psi
Readability Allowance	+22 psi
T/S Limit	2194 psig

This value is supported in Attachment 4, Table 6.1 and Appendix A.

The proposed T/S lower pressure limit is set conservatively at 2200 psig. This is the value we plan to submit for Unit 1 in the future and was selected to make the T/Ss of the two units more consistent.

i) F_p Penalty, T/S 3/4.2.6

This change is numbered 049 in Attachment 3. The allowable power level (APL) limiting condition for operation will be supported in Cycle 8 by Part B of WCAP 10217-A. This results in a change in the penalty for increasing peaking factors. The current specification requires monitoring $F^N_{\Delta H}$ for increases in successive steady state flux maps. This submittal proposes to monitor instead the maximum over Z of $\{F_Q(Z)/K(Z)\}$. This change is based on p. B-4 of WCAP 10217-A. Part B of WCAP 10217-A is included in this submittal as Attachment 10 to facilitate the reviewer's task.

j) Pressurizer Level Reactor Trip Response Time, T/S Table 3.3-2

This change is numbered 050 in Attachment 3. This protection function is needed to prevent pressurizer fill for certain cases of uncontrolled control rod assembly bank withdrawal at power. This is discussed in Sections B.3.2A.2, B.3.2A.3, and B.3.2A.4 of Attachment 4, Appendix B and is supported by Table 6.1 and Appendix A of Attachment 4.

k) Changes to Protection Response Times.

These changes are numbered 051 and 054 in Attachment 3. We propose to relax six protection response times listed in T/S Table 3.3-2 from present values to values used in the most recent analyses. The proposed changes are:

Functional Unit	Present Response Time	Proposed Response Time
Pressurizer Pressure - Low	1.0 sec	2.0 sec
Pressurizer Pressure - High	1.0 sec	2.0 sec
Loss of Flow-Single Loop	0.6 sec	1.0 sec
Loss of Flow-Two Loops	0.6 sec	1.0 sec
Steam Generator Water Level Low-Low	1.5 sec	2.0 sec
Undervoltage - Reactor Coolant Pumps	1.2 sec	1.5 sec

The new values are listed in Table B.2-2 and C.3.1-3 of Appendix B of Attachment 4 and are supported by Table 6.1 and Appendix A of Attachment 4.

l) High Steam Flow Setpoints, T/S Table 3.3-4, Functional Unit 4.d

This change is numbered 066 in Attachment 3. New values for high steam flow setpoints are proposed. This functional unit was not assumed in any safety analyses for Cook Nuclear Plant Unit 2. The setpoints are based on the mass and energy release inside containment which conservatively assumed safeguards actuation on low steam line pressure coincident with high steam flow in order to bound Cook Nuclear Plant Unit 1. This analysis is described in Attachment 5, particularly page S-3.3-11. The setpoints are provided in Attachment 4, Table 6.1 and Appendix A.

m) Response Times for Containment Pressure-High Safeguards Actuation, T/S Table 3.3-5, Initiating Signal 2

These changes are numbered 069, 070, and 071 in Attachment 3. The addition of response times for Engineered Safety Features Actuation on containment pressure high are proposed based on the mass and energy releases inside containment. This analysis is discussed in WCAP 11902, Supplement 1. The description is included in this submission as Attachment 5. The safeguards employed for these transients are discussed on page S-3.3-12 of that attachment. The response times are provided in Attachment 4, Table 6.1 and Appendix A.

n) Response Times for Steam Generator Water Level - High-High Safeguards Actuation, T/S Table 3.3-5, Initiating Signal 8

These changes are numbered 072 and 073 in Attachment 3. This safeguards function is required to terminate main feedwater flow in events where the main feedwater system malfunctions causing an increase in feedwater flow. This is discussed in Section B.3.8A.2.3 of Appendix B to Attachment 4. The response times are provided in Table B.2-3 of Appendix B and in Appendix A of Attachment 4.

o) Pressure and Volume Limits for Accumulators, T/S 3/4.5.1

These changes are numbered 084 and 085 in Attachment 3.
New values for the accumulator volume and pressure ranges are proposed. They are based on the values used in the new LOCA analyses. The values used in the analyses are indicated in Tables C.3.1-2 and C.3.2-1 of Attachment 4, Appendix C. The applicability of the analyses to mixed core configuration is discussed in Attachment 4 Section 5.2 and in Attachment 4, Appendix C. The limit values for the pressure and volume limits are found in Attachment 4, Table 6.1 and Appendix A.

p) Safety Injection and RHR Pump Degradation, T/S 4.5.2.f

These changes are numbered 086 and 087 in Attachment 3. AEP proposes new surveillance acceptance criteria for the safety injection (SI) and residual heat removal (RHR) pumps. These changes are based on the analyses performed for Unit 2 Cycle 8 and the analyses performed for both units as a part of the reduced temperature and pressure and the uprate programs. The section of WCAP 11908, "Containment Integrity Analysis" which describes the major analytic assumptions is included as Attachment 11 for the reviewer's convenience. WCAP 11908 was submitted on August 22, 1988 in AEP:NRC:1024D. The differential pressures, calculated by our contractor Westinghouse, are found in Attachment 4, Table 6.1 and Appendix A.

These differential pressures are converted to discharge pressures as follows:

	SI Pump	RHR Pump
Differential Pressure	1385 psid	160 psid
Suction Pressure	24 psi	30 psi
Discharge Pressure	1409 psig	190 psig

The suction pressure values are conservatively based on the maximum elevation of water possible in the refueling water storage tank (RWST) (limited by the elevation of the tank overflow piping), and the T/S minimum RWST tank temperature.

We are not proposing to change the discharge pressure for the centrifugal charging pump because our work on the new mass and energy releases analysis outside containment is incomplete.

However, we propose to revise the Unit 1 discharge pressures for the RHR and SI pumps. The present and proposed T/S minimum discharge pressure requirements are as follows:

Pump	Present Requirement (psig)	Proposed Requirements (psig)
SI	1345	1409
RHR	165	190

The revised requirements result from our discovery that the T/S requirements, which were recently approved (Amendment 126) for the Unit 1 Reduced Temperature and Pressure (RTP) Program, are inconsistent with the assumptions used by Westinghouse in performing the RTP analyses.

In WCAP 11902, which contains the RTP analyses, Westinghouse specified the required pump pressures as 1345 psi differential for the SI pumps and 165 psi differential for the RHR pumps. The Cook Nuclear Plant T/Ss are written in terms of pump discharge pressure, however, and the conversion from differential pressure to discharge pressure was inadvertently neglected. In addition, Westinghouse recently informed us that the value of 1345 psid differential pressure that was supplied in WCAP 11902 was in error and should actually be 1385 psi differential.

The correct differential pressures are found in WCAP 11902, Supplement Table S-3.13-2 and are identical to the values applicable to Unit 2. Therefore, the correct values for the Unit 1 discharge pressures are also identical to those calculated above for Unit 2. Table S-3.13-2 is included in Attachment 5 for the reviewer's convenience.

Upon discovery of the discrepancies, the more stringent requirements for the SI and RHR pumps were implemented administratively. Surveillance test results have demonstrated that the pumps met the more stringent requirements at all times since the issuance of Amendment 126 and therefore the discrepancies did not impact safety.

q) Boron Injection System Flow Imbalance, T/S 4.5.2.h

This change is numbered 089 in Attachment 3. SBLOCA was analyzed assuming a SI flow consistent with the proposed T/Ss. This is discussed in Attachment 4, Appendix C, Section C.3.2. The other analyses are not significantly affected by the proposed T/S. This parameter is also provided in Table 6.1 and Appendix A of Attachment 4.

<u>10 CFR 50.92 Evaluation for T/S Change Group 1</u>

Per 10 CFR 50.92, a proposed amendment will not involve a significant hazards consideration if the proposed amendment does not:

1) Involve a significant increase in the probability or consequences of an accident previously analyzed,

2) Create the possibility of a new or different kind of accident from any accident previously analyzed or evaluated, or

3) Involve a significant reduction in a margin of safety.

Criterion 1

The proposed T/S changes to support Cycle 8 operations are accompanied by extensive analyses and evaluations which indicate that they will not result in an unsafe condition at the plant. The analyses and evaluations support our conclusion that the proposed T/S changes will not involve a significant increase in the probability or consequences of any accident previously analyzed.

Criterion 2

The Cycle 8 analyses and evaluations comply with the licensing basis of the plant. Thus, the proposed T/S changes should not create the possibility of a new or different kind of accident from any previously analyzed or evaluated.

Criterion 3

The proposed T/S changes to support Cycle 8 operations are accompanied by extensive analyses and evaluations which indicate that they will not result in an unsafe condition at the plant. The analyses and evaluations support our conclusion that the proposed T/S changes will not involve a significant reduction in any margin of safety.

The conclusions which our contractor, Westinghouse, drew from the safety evaluations and analyses are found on page 4 of Attachment 4. Westinghouse in part based their conclusions on the implementation of their proposed T/S changes in Table 6.1 and Appendix A of Attachment 4. We have proposed those changes unless the change is in another submittal, a more conservative change is proposed, or no change is more conservative than the Westinghouse proposal. Therefore, the Westinghouse conclusions support our 10 CFR 50.92 analysis.

Lastly, we note that the Commission has provided guidance concerning the determination of significant hazards by providing examples (48 FR 14870) of amendments considered not likely to involve a significant hazards consideration. The sixth of these examples refers to changes that may result in some increase to the probability or consequences of a previously analyzed accident, but the results of which are within limits established as acceptable.

The analyses performed by Westinghouse in support of Unit 2 Cycle 8 operation comply with the licensing basis of the unit. Thus, we believe the example cited is applicable and that the changes should not involve a significant hazards consideration.

2) <u>T/S Change Group 2</u>

Removal of transition mode T/Ss associated with the Unit 2 Cycle 6 reanalysis.

a) Background

In our March 14, 1986, submittal to the NRC (AEP:NRC:0916I) containing the proposed T/S changes

for the Unit 2 Cycle 6 reload, we included several T/Ss which are not in the Westinghouse Standard T/Ss (STS). They were incorporated in the Unit 2 T/Ss by Amendment 82 to Operating License DPR-74. These T/Ss resulted from a review of abnormal operating occurrences (AOO) and postulated accidents (PA) in the transition, shutdown, and refueling modes of operation (Modes 2,3,4,5 and 6). Reviews of AOOs and PAs in Modes 2 through 6 were conducted in conjunction with analyses and evaluations performed by Exxon Nuclear Company (now Advanced Nuclear Fuels Corporation) at the request of the NRC.

One non-standard T/S was proposed for Unit 1 in our submittal AEP:NRC:0916W. It is the shutdown margin in shutdown Modes 4 and 5. This was included in the Unit 1 T/Ss by Amendment 120 to Operating License DPR-58. We propose to revise the Unit 1 shutdown margin T/Ss in the same fashion that we propose to revise the Unit 2 shutdown margin T/Ss.

AEP proposes to remove a number of these requirements from the T/Ss, thereby creating a structure more like that of the STS. The issues that are addressed by the T/Ss which we propose to delete will be addressed in the future by appropriate administrative controls. We believe this will simplify our T/Ss in conformance with industry practice and ensure the continued safe operation of the Cook Nuclear Plant.

Our plan was discussed with the NRC staff in a meeting on June 12, 1989 at NRC headquarters on Rockville Pike in Maryland. The staff expressed no concern with this aspect of our discussion. A summary of meeting topics and issues of concern to the staff are documented in a letter to M. P. Alexich, Vice President of Indiana Michigan Power Company with its attachments from Joseph G. Giitter, Project Manager of the NRC staff, dated August 3, 1989. For the convenience of the reviewer, this letter is included in this submittal as Attachment 9.

The T/Ss which are impacted by the proposed changes in this group are:

(1)	3/4.1.1.1 (Unit 1, Unit 2)	Shutdown Margin, Standby, Startup, and Power Operations
(2)	4.1.1.1.3 (Unit 2)	Shutdown Margin Surveillance

(3) 3/4.1.1.2 (Unit 1, Unit 2) Shutdown Margin, Shutdown

(4) 3/4.2.5.2 (Unit 2) DNB Parameters, Modes 2,3,4,5

(5) Table 3.3-3 (Unit 2) ESF Actuation System Instrumentation

b) Shutdown Margin in Shutdown Modes 4 and 5

These changes are numbered 027, 032 and 033 in Attachment 3. AEP proposes to rearrange T/Ss 3/4.1.1.1 and 3/4.1.1.2 into STS format by moving the Mode 4 requirement from 3.1.1.2 to 3.1.1.1 and deleting Figure 3.1-3 and related references. In the Cycle 6 submittal the Mode 4 shutdown margin was moved from 3/4.1.1.1 to 3/4.1.1.2. The Cycle 6 change to T/S 3/4.1.1.2, which then included Mode 4 and Mode 5 in the same T/S, was a revision that reflected the results of the boron dilution accident analysis when heat removal is via the residual heat removal (RHR) system. This analysis was performed by ANF.

AEP proposes to return 3/4.1.1.1 and 3/4.1.1.2 to STS format which will be based on SDM requirements for analyses of the plant in Modes 1 and 2. Protection for boron dilution accidents when the plant is operated on RHR will be addressed by administrative controls. This proposal applies to both units.

AEP presently plans to use the Westinghouse developed methods to ensure adequate operator response time in the event that either unit is subject to a dilution incident when operating in the various RHR cooling configurations. The methodology will address the limiting case of operation at reduced coolant inventory in the primary system. This methodology is a modification of Attachment 1 of the July 8, 1980, letter from T. M. Anderson of Westinghouse Electric Corporation to Victor Stello of the NRC. The identifier of Mr. Anderson's letter is NS-TMS-2273. The modified methodology is similar to the original methodology. The NS-TMA-2273 methodology, or a

similar methodology which is currently reflected in T/S 3/4.1.1.2 of Cook Nuclear Plant Units 1 and 2, has been in use on Unit 1 since the beginning of Cycle 6 and on Unit 2 since the beginning of Cycle 3.

The RHR function is a generic function. STS T/S 3.1.1.2 does not include unique SDM requirements for RHR operation. Furthermore, adequate response time for the dilution incident when cooling on RHR was ensured by administrative controls prior to Cycle 6 of Unit 2 and Cycle 10 of Unit 1. AEP will continue to ensure adequate operator response time by methodologies discussed above, or other effective methodologies.

c) Shutdown Margin Surveillance with ESF Actuations Blocked in Mode 3

These changes are numbered 031 and 058 in Attachment 3. The surveillance requirement 4.1.1.1.3 in the currently approved T/Ss is to be removed to further bring the Unit 2 T/Ss into conformance with the STS. AEP also proposes to revise footnotes # and ## of Table 3.3-3. These footnotes are currently designed to be used with surveillance 4.1.1.1.3. The two footnotes, # and ##, will be returned to their original content, which is consistent with the STS.

Surveillance 4.1.1.1.3 and the elaboration of footnotes # and ## were proposed as part of AEP:NRC:0916I for Cycle 6. As was indicated in XN-NF-85-28 (P), Disposition of Standard Review Plan Chapter 15 Events, these requirements were instituted to protect against a failure in the steam dump to condenser system below P-12. They were designed to ensure that in the event of a steam dump to condenser failure, the core would not become critical. A single failure in the steam dump to condenser below P-12 could potentially open three steam dump valves. These valves would remain open until the operator was able to take action. Since this event is symmetric, ESF actuation on differential pressure between steam lines would not be expected. Safeguards actuations that would protect against symmetric events are blocked at P-11 and P-12.

Blocking of safeguards actuation on P-11 and P-12 are generic functions. These actions are required to cooldown and depressurize the units without a consequential safeguards actuation. The bases of the setpoints are generic. P-12 permits blocking of safeguards actuation on low secondary pressure when

Tavg is below 541°F. P-11 is above the low pressure SI setpoint and permits blocking of safeguards actuation on low pressurizer pressure. Blocking safeguards avoids undesired actuations during heatup and cooldown.

To protect against increased steam loads with safeguards blocked below P-11 and P-12, AEP plans to ensure cold shutdown SDM is available prior to blocking safeguards. Under certain circumstances in which cooldown is urgent, verification of SDM may not be complete prior to initializing of cooldown. An example of such a circumstance is a steam generator tube leak. Although safeguards will not be actuated by a tube leak, it is nevertheless important to promptly cooldown and depressurize. Procedures have been implemented for both units to address this possibility.

As noted above, the STS do not include surveillance 4.1.1.1.3 or the elaborations of footnotes # and ##. AEP will ensure that an increase in steam load below P-11 and P-12 will not result in re-criticality by the methods discussed above or by other effective methods.

d) DNB Parameters - Modes 2, 3, 4 and 5

These changes are numbered 047 and 048 in Attachment 3. AEP proposes to delete the Modes 2, 3, 4 and 5 DNB requirements of T/S 3/4.2.5.2. AEP also proposes to remove Table 3.2-2 which is associated with specification 3.2.5.2. This specification was proposed as part of AEP:NRC:0916I for Cycle 6. The intent of this T/S is to prevent DNB from occurring should an uncontrolled rod withdrawal accident from subcritical occur.

AEP believes that other T/S requirements, which will be left in place, in conjunction with administrative controls are adequate to ensure that the assumptions of the analysis are met. T/Ss 3.4.1.2.c and 3.4.1.3.c which require three reactor coolant loops in operation when control rods are capable of withdrawal will be left in place. This provision is conservative relative to the analysis that was performed by Westinghouse for this submission which required only two operating reactor coolant loops. T/S 3.4.1.1 requires that all reactor coolant pumps be operable in Modes 1 and 2. Administrative controls require all pumps when operating above 541°F. Therefore, in practice all pumps are in operation during startup. The operability

requirements of T/S Table 3.3-1 for the nuclear instrumentation high flux low setpoint reactor trip will also be left in place. In addition, the non-safety grade intermediate range and source range trips are required to be operable by T/S Table 3.3-1 when control rods are capable of withdrawal. Therefore, the source range high neutron flux trip will be available to terminate the event by tripping any withdrawn and withdrawing rods before any significant power level could be attained. With the source range trip effective, DNB and primary system flow rate need not be considered. Also, the reactivity insertion rate would be slower when in any of the subcritical modes because the rod control system in Modes 2-5 is expected to be in bank select. Therefore, a single failure in the rod control system could cause the withdrawal of only one bank, and its withdrawal rate would be expected to be slower than the maximum rod speed that is possible when in automatic rod control (and is assumed in the UFSAR analysis).

Administrative controls will ensure that the unit will be sufficiently shutdown such that criticality cannot occur with any bank fully withdrawn if the control rods are capable of withdrawal and the reactor is not at hot zero power (HZP) conditions. Administrative controls also ensure that the upper temperature is limited at the HZP conditions by the setpoint of the non-safety grade steam generator power operated relief valves. The setpoint for these valves is 1040 psia limiting RCS temperature to approximately 549°F.

The analysis performed for the Donald C. Cook Nuclear Plant is similar to that of other plants. Furthermore, the generic STS do not include DNB specifications for Modes 2, 3, 4 and 5. AEP believes that the requirements and controls discussed above will ensure that an inadvertent criticality will not occur.

e) Footnote $ to Applicable Mode for Turbine Trip and Feedwater Isolation in Table 3.3-3 and Table 4.3.2

These changes are numbered 057, 059, 078 and 082 in Attachment 3. The $ footnote was added to the Applicable Modes column of T/S Table 3.3-3 and the Modes in Which Surveillance Required of Table 4.3-2 as result of an evaluation of the increased feedwater event in Mode 4 performed by Exxon Nuclear Company (now Advanced Nuclear Fuels Corporation, ANF). AEP proposes to delete this footnote from Tables 3.3-3 and 4.3-2. The ANF calculation requires the steam

generator water level--high-high functional unit to be operable when operating with the main feedpumps feeding the steam generators in Modes 3 and 4. The availability of feedwater isolation on high-high steam generator level limits the volume of cold water that can be added to the steam generators in any feedwater malfunction. This limits reactivity addition to the core.

Preheated feedwater is used frequently in Mode 3 operation. AEP plans to leave the Mode 3 requirement in place. The Mode 3 requirement has historically been in the T/Ss for both Cook Nuclear Plant units although it is not included in STS, Rev. 4. However, the likelihood of operating the main feedwater system in any mode other than 1, 2 or 3 is extremely remote. The circumstances associated with this contingency will be addressed administratively as needed.

f) <u>10 CFR 50.92 Evaluation for T/S Change Group 2</u>

Per 10 CFR 50.92, a proposed amendment to an operating license will not involve a significant hazards if the proposed amendment satisfies the following criteria:

1) Does not involve a significant increase in the probability or consequence of an accident previously analyzed,

2) Does not create the possibility of a new or different kind of accident from any accident previously analyzed or evaluated, or

3) Does not involve a significant reduction in a margin of safety.

Criterion 1

The proposed T/S changes do not involve a physical change to the plant. The procedures and administrative controls for the plant will either remain in place as described above, or in some cases be replaced by controls which we believe are of comparable effectiveness. Therefore, we conclude that the proposed T/S changes will not result in a significant increase in the probability or consequences of any accident previously analyzed.

Criterion 2

The proposed T/S changes do not involve a physical change to the plant. The procedures and administrative controls for the plant will either remain in place as described above, or in some cases be replaced by controls which we believe are of comparable effectiveness. Therefore, we conclude that the proposed changes will not create the possibility of a new or different kind of accident from any previously evaluated.

Criterion 3

The proposed T/S changes do not involve a physical change to the plant. The procedures and administrative controls for the plant will either remain in place as described above or, in some cases, be replaced by controls which we believe are of comparable effectiveness. Therefore, we conclude that the proposed T/S changes will not involve a significant reduction in any margin of safety.

Lastly, we note that the Commission has provided guidance concerning the determination of significant hazards by providing examples (48 FR 14870) of amendments considered not likely to involve significant hazards consideration. The sixth of these refers to changes that may result in some increase to the probability or consequences of a previously analyzed accident, but the results of which are within limits established as acceptable.

The proposed T/S changes in Change Group 2 remove certain activities from T/S control. However, since operation of the plant will be governed by the continuation of existing controls or controls of comparable effectiveness, the results of the underlying evaluation should remain within limits established as acceptable. For this reason we believe the sixth example bounds the proposed Change Group 2 changes.

3) <u>T/S Change Group 3</u>

Changes to Unit 2 T/Ss for consistency with Unit 1 T/Ss

a) Discussion

There is one change in this category. This change is numbered 083 in Attachment 3. It is a proposal to increase the water level for operability in the

pressurizer. This change was submitted for Unit 1 in our reduced temperature and pressure (RTP) program submittal, AEP:NRC:1067 dated October 14, 1988. It was approved in Amendment No. 126 to Operating Licensing DPR-58.

We propose to increase the maximum pressurizer level to 92% of span. The purpose of the maximum pressurizer level limit, as described in the Bases, is to ensure that a bubble can exist in the pressurizer. Westinghouse has determined that a bubble can be maintained at the 92% level. The change is described in detail in Section 3.13 of WCAP 11902 which is included in this submittal as Attachment 6 for the reviewer's convenience. The change will allow operational flexibility at the higher end of the Tavg spectrum analyzed for the RTP program.

b) <u>10 CFR 50.92 Evaluation for T/S Change Group 3</u>

Per 10 CFR 50.92, a proposed amendment will not involve a significant hazards consideration if the proposed amendment does not:

1) Involve a significant increase in the probability or consequences of an accident previously analyzed,

2) Create the possibility of a new or different kind of accident from any previously analyzed or evaluated, or

3) Involve a significant reduction in a margin of safety.

Criterion 1

The proposed T/S change proposed is accompanied by an evaluation which indicates that it will not result in an unsafe condition at the plant. The evaluations support our conclusion that the change, which has already been approved for Unit 1, will not involve a significant increase in the probability or consequences of any accident previously analyzed.

Criterion 2

The evaluation of the proposed T/S change complies with the licensing basis of the plant. Thus, this change should not create the possibility of a new or different kind of accident from any previously analyzed or evaluated.

Criterion 3

The proposed T/S change is accompanied by an evaluation which indicates that it will not result in an unsafe condition at the plant. The evaluation supports our conclusion that the proposed T/S change, which has already been approved for Unit 1, will not involve a significant reduction in any margin of safety.

Lastly, we note that the Commission has provided guidance concerning the determination of significant hazards by providing examples (48 FR 14870) of amendments considered not likely to involve significant hazards consideration. The sixth of these examples refers to changes that may result in some increase to the probability or consequences of a previously analyzed accident, but the results of which are within limits established as acceptable.

The evaluation for this proposed T/S change complies with the licensing basis of the plant. Thus, we believe the example cited is applicable and that the change should not involve a significant hazards consideration.

4) <u>T/S Change Group 4</u>

a) Discussion

This T/S change group consists of changes that are purely editorial in nature. The Attachment 3 identification numbers for these proposed changes are:

<u>Unit 1</u>
116, 118, 123

<u>Unit 2</u>
003, 004, 005, 006, 009, 010, 017, 018, 024, 025, 028, 029, 030, 035, 037, 038, 040, 044, 045, 046, 052, 053, 055, 056, 060, 061, 062, 063, 064, 065, 067, 068, 074, 075, 076, 077, 079, 080, 081, 088, 090.

The changes in this group include proposals to enhance the readability of the T/Ss, to move existing text, or to perform other non-substantive changes as described in Attachment 3.

b) 10 CFR 50.92 Evaluation for T/S Change Group 4

Per 10 CFR 50.92, a proposed amendment will involve a no significant hazards consideration if the proposed amendment does not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated,

(2) Create the possibility of a new or different kind of accident from any accident previously analyzed or evaluated, or

(3) Involve a significant reduction in a margin of safety.

Criterion 1

These proposed T/S changes, being editorial in nature and intended to improve the readability of the T/Ss, will not reduce in any way requirements or commitments in the existing T/Ss. Thus, we believe that no increase in the probability or consequences of a previously evaluated accident would be expected as a result of these proposed T/S changes.

Criterion 2

We believe that these purely editorial proposed T/S changes will not create the possibility of a new or different kind of accident from any previously evaluated, because no change to the plant or plant operations will result.

Criterion 3

We believe that the proposed T/S changes will not involve a significant reduction in any margin of safety, because all accident analyses and nuclear design bases remain unchanged.

Lastly, we note that the Commission has provided guidance concerning the determination of significant hazards by providing certain examples (48 FR 14870) of amendments considered not likely to involve significant hazards consideration. The first of these examples refers to changes that are purely administrative in nature: for example, changes to achieve consistency throughout the T/Ss, correction of an error, or a change in nomenclature. This group of proposed changes is intended to achieve consistency between the Unit 1 and 2 T/Ss, or to improve the overall readability of the T/S document. As these changes are purely editorial and do not impact safety in any way, we believe the Federal Register example cited is applicable and that the changes involve no significant hazards consideration.

5) T/S Change Group 5

Changes to the surveillance requirement for power distribution limits/axial flux difference for Units 1 and 2.

(a) Discussion

This change is numbered 039 in Attachment 3. The proposed change will eliminate unnecessary surveillances of the indicated axial flux difference (AFD) during power operation above 15% of rated thermal power. The current T/Ss for both units require in 4.2.1.1.a.2 that the indicated AFD for each operable excore channel be monitored "at least once per hour for the first 24 hours after restoring the AFD monitor alarm to operable status." The proposed T/S change will add a caveat. This surveillance will only be required "if the AFD had been outside of the target band for any period of time in the previous 24 hours of operation."

The requirement to monitor the indicated AFD for 24 hours after restoring the AFD monitor alarm to operable status results from the fact that the current computer program that monitors the AFD has no provision for updating the cumulative time out of the target band. The cumulative time out of the target band is reset to zero when the AFD monitor alarm is restored to operable status. In the case where it can be demonstrated that the AFD indeed has not been outside the target band in the previous 24-hour period, zero is the proper time for cumulative

time out of the target band. In this case, no further monitoring is needed. In the other case, where the AFD has been out of the target band in the previous 24 hours, zero is non-conservative and is not acceptable for the cumulative time out of the target band. Therefore, the AFD must be manually monitored for 24 hours to ensure that sometime in the first 24 hours of operation that limit on cumulative time out of the target band is not exceeded. After 24 hours, the rolling log of cumulative time out of target band will start to overwrite itself and complete information for the previous 24 hours is contained in the log.

b) <u>10 CFR 50.92 Evaluation for T/S Change Group 5</u>

Per 10 CFR 50.92, a proposed amendment will not involve a significant hazards consideration if the proposed amendment does not:

1) Involve a significant increase in the probability or consequences of an accident previously analyzed,

2) Create the possibility of a new or different kind of accident from any previously analyzed or evaluated, or

3) Involve a significant reduction in a margin of safety.

Criterion 1

This proposed T/S change will not result in an increase in the probability or consequences of an accident previously analyzed. The requirement to monitor the AFD for the first 24 hours after restoring the AFD to operable status will be eliminated only when the axial power distribution has been within the target band for 24 hours prior to restoring the alarm to operable status. This will not impact the safety of the plant because the alarm can accurately monitor the cumulative time out of the target band in this case. If the AFD has been outside of the target band at any time in the 24 hours prior to restoring the alarm to operable status, then the surveillance of monitoring the AFD for the first 24 hours is still required.

Criterion 2

We believe that adding the caveat to the additional surveillance requirement will not result in a new or different kind of accident from any previously evaluated.

The modified surveillance will ensure that the cumulative time out of the target band will be conservatively calculated for all circumstances either by the plant process computer or by manual monitoring.

Criterion 3

The requirement to monitor the AFD for the first 24 hours after restoring the AFD to operable status will be eliminated only when the axial power distribution has been within the target band for 24 hours prior to restoring the alarm to operable status. This will not impact the safety of the plant because the alarm can accurately monitor the cumulative time out of the target band in this case. If the AFD has been outside of the target band at any time in the 24 hours prior to restoring the alarm to operable status, then the surveillance of monitoring the AFD for the first 24 hours is still required. We conclude that this proposed T/S change will not involve a significant reduction in any margin of safety.

Lastly, we note that the Commission has provided guidance concerning the determination of significant hazards by providing examples (48 FR 14870) of amendments considered not likely to involve significant hazards consideration. The sixth of these refers to changes that may result in some increase to the probability or consequences of a previously analyzed accident, but the results of which are within limits established as acceptable.

We believe this example is applicable to the proposed Change Group 5 T/S change. As indicated above, we anticipate no consequential impact on delta flux monitoring and control to result from this change. Therefore, the underlying evaluations and analyses for delta flux control should clearly remain within limits established as acceptable.

6) T/S Change Group 6

Changes to Unit 1 T/Ss for consistency with Unit 2 T/Ss. These changes are numbered 115, 117, 119, 120, 121 and 122. The corresponding Unit 2 changes are numbered 027, 032, 033, 039, 086 and 087 in Attachment 3.

These changes to the Unit 1 T/Ss are being proposed to make the Unit 1 T/Ss more like the Unit 2 T/Ss. The particular Unit 1 changes being proposed have an identical or essentially identical justification for both units and the Unit 2 justification is part of this submittal. The proposal to remove special shutdown margin requirements when operating on RHR, T/S Sections 3.1.1.1 and 3.1.1.2 and Figure 3.1-3, discussed in T/S Change Group 2b, is proposed for both units. The modification to the axial flux difference (AFD) surveillance 4.2.1.1.a.2, discussed in T/S Change Group 5 is proposed for both units. Finally, the new acceptance criteria for safety injection and residual heat removal pumps discussed in T/S Change Group 1p is proposed for both units for T/S Section 4.5.2.

A 10 CFR 50.92 evaluation is not included for Change Group 6 because we believe the evaluations performed for the corresponding Unit 2 changes in T/S Change Groups 1p, 2b, and 5 apply to both Unit 1 and Unit 2.

IV. Proposed Changes to the Bases

In addition to the changes to the T/Ss described above, we have also proposed changes to the Bases section to reflect both changes in the safety analyses and changes in the T/Ss. Descriptions of these changes have been included in Attachment 3.

V. Conclusion

We believe that the proposed changes do not involve a significant hazards consideration because operation of Cook Nuclear Plant Unit 2 in accordance with these changes would not:

(1) Involve a significant increase in the probability of occurrence or consequences of an accident previously analyzed,

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated, or

(3) Involve a significant reduction in a margin of safety.

This conclusion is based on our evaluation of the changes, which has determined that all proposed changes that are not administrative in nature, consistent with the STS, or consistent with the design basis of the plant are clearly traceable to the various supporting safety analyses, as referenced by Attachment 3. Assuming Commission acceptance of these analyses, it is our belief that they successfully demonstrate that applicable safety limits and margins to safety will be maintained.